NUCLEAR SAFETY

A REVIEW OF RECENT DEVELOPMENTS

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foreword

This quarterly journal is a Technical Progress Review prepared by the Oak Ridge National Laboratory at the request of the Office of Technical Information, U. S. Atomic Energy Commission. This Review is intended to assist those interested in keeping abreast of significant developments in the field of nuclear safety. *Nuclear Safety* is not a comprehensive abstract of all literature published in this field during a given quarter; rather it is a mechanism for presenting concise reviews of selected subjects as prevailing interest and available information warrant.

Coverage of the Review is limited to topics relevant to the analysis and control of hazards associated with nuclear reactors, operations involving fissionable materials, and the products of nuclear fission. Primary emphasis is on safety in reactor design, construction, and operation; however, safety considerations in reactor fuel fabrication, spent-fuel processing, nuclear waste disposal, and related operations are also treated. Safety in the use of radioisotopes in industry, medicine, and research is excluded, as are most topics considered to be in the province of health physics. Even with these exclusions, nuclear safety overlaps such diverse fields as nuclear physics, solid-state physics, mechanics, chemistry, meteorology, geology, seismology, metallurgy, law, and nearly all branches of engineering. The authors will therefore review material from these fields which, in their opinion, has a direct bearing on nuclear safety.

Two distinctly different types of articles are in this issue of *Nuclear Safety:* reviews of current literature and special review articles on specific topics. The editors feel that each of these articles makes a necessary and distinctive contribution to this journal. The special review articles permit discussion of pertinent subjects which cannot be adequately considered by reference to only the current literature. The current review articles, however, constitute the major portion of this issue.

All incoming literature (including reports, books, American and foreign technical journals, and transactions) is examined for subjects within our area of interest. This material is collected, grouped, and reviewed by experts. Interpretations, in any article, represent the opinions of the editors, who are employees of the Oak Ridge National Laboratory. Readers are urged to consult the references to original work for more complete information.

In addition to the invited contributors, many members of the Oak Ridge National Laboratory staff wrote review material, reviewed manuscripts, or otherwise contributed to this publication. Their contributions are gratefully acknowledged.

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Section I

GENERAL SAFETY CRITERIA

Biological Basis for Routine Exposure Criteria

One of the major limiting considerations in the design and operation of nuclear reactors, particle accelerators, and other radiation sources is concern for human safety. This concern has brought about the development of protection standards to safeguard against radiation injury. The biological criteria for these standards are reviewed briefly here. More extensive discussions of the subject have appeared elsewhere.[1,2]

Although it was evident within months after Roentgen discovered X rays in 1895 that overexposure to these rays damaged the skin and deeper tissues, it was not recognized until several years later that occupational exposure to successive small doses might lead insidiously to irreversible effects, such as cancer. A timetable of radiation injury reports adapted from the historical notes of Stone[3] is presented in Table I-1.

Table I-1 RADIATION INJURY REPORTS FOLLOWING DISCOVERY OF X RAYS BY ROENTGEN IN 1895

Date	Type of injury	Reported by
1896	Dermatitis of hands	Grubbé
1896	Smarting of eyes	Edison
1896	Epilation	Daniel
1897	Constitutional symptoms	Walsh
1899	Degeneration of blood vessels	Gassman
1902	Cancer in X-ray ulcer	Frieben
1903	Bone growth inhibited	Perthes
1903	Sterilization produced	Albers-Schonberg
1904	Blood changes produced	Milchner and Mosse
1906	Bone-marrow changes demonstrated	Warthin
1911	Leukemia in five radiation workers	Jagie
1912	Anemia in two X-ray workers	Béclère

Although some need for protection was apparent almost from the beginning, attempts to define permissible exposure limits were slow in developing, as indicated in the following tabulation, which was also adapted from the work of Stone:[3]

1902 Rollins established photographic indication of "safe" intensity

1921 British X-Ray and Radium Protection Committee considered "establishing a maximum tolerance dose"

1925 First attempt was made by Mutscheller to define a "tolerance dose"

1931 U. S. Advisory Committee on X-Ray and Radium Protection recommended a tolerance dose of 0.2 r/day

1934 International X-Ray and Radium Committee endorsed tolerance dose of 0.2 r/day

1936 U. S. Advisory Committee on X-Ray and Radium Protection reduced tolerance dose to 0.1 r/day

1942 Metallurgical (Plutonium) Project adopted tolerance dose of 0.1 r/day

1950 International Commission on Radiological Protection recommended 0.3 r/week (measured in air) = 0.05 r/day

1956 International Commission on Radiological Protection recommended 5 r/year

In 1921, the newly formed British X-Ray and Radium Protection Committee considered establishing a maximum tolerance dose to be defined in terms of a reproducible biological standard and expressed, if possible, in physical units.[3] In 1928, the International Commission

on Radiological Protection (ICRP) was established to cope with the same problem. In accordance with its recommendations, the Advisory Committee on X-Ray and Radium Protection was formed the following year in the United States under the sponsorship of the National Bureau of Standards and with the cooperation of leading radiological organizations. In 1931, this committee recommended a tolerance dose of 0.2 r/day (measured in air), which was endorsed in 1934 by the ICRP. Two years later, however, the U. S. committee reduced its recommended tolerance level to 0.1 r/day, apparently for two main reasons. First, the reduction was made to conform more closely to the tolerance level in Europe, where the dose was customarily measured on the skin, thus including backscatter, rather than in air as was customary in the United States. Second, the reduction was intended to compensate for the increasing prevalence of 200-kv X-ray machines and the resulting increase in the percentage of the surface dose penetrating to deeper parts of the body.[3]

Selection of the tolerance level of 0.1 r/day, as opposed to higher or lower levels, was based on study of the occupational exposures of people who had worked with radiation for many years without detectable injury. The number of such people was relatively small, however, and the estimation of their exposure levels was highly uncertain. Hence, with the advent of nuclear fission and the greatly increased risks, the basis for this tolerance dose no longer seemed adequate. It became increasingly important, therefore, to establish a more reliable tolerance dose without unnecessarily restricting the development of atomic energy and allied fields. The stepwise lowering of the maximum permissible dose in 1948 and 1956, as indicated in Table I-2 (adapted from reference 4), occurred not because of newly documented evidence of injury at the former tolerance level but because of new knowledge about the effects of low-level radiation which dictated a need for greater margins of safety. The biological considerations underlying these developments are surveyed here for injuries affecting the exposed individual (somatic effects), his descendants (genetic effects), or both.

Table I-2 HISTORICAL SEQUENCE OF REDUCTIONS IN MAXIMUM PERMISSIBLE RADIATION DOSES

Exposure period	Maximum permissible doses, r			
	1931–36	1936–48	1948–56	1956
Per day	0.2	0.1	0.05	
Per 6-day week	1.2	0.6	0.3	0.3
Per 50-week year	60	30	15	5
Per decade	600	300	150	50
Up to age 60	2400	1200	600	200

Somatic Effects of Radiation

Somatic effects may be grouped into two major categories: (1) early, i.e., developing soon after irradiation (within days or weeks); and (2) late, i.e., developing only after a long latent period (months or years). The early somatic effects result, in general, from the destruction of radiosensitive cells. Although promptly visible degenerative changes may be induced in certain circulating blood cells by less than 1 rad (reference 5) and in certain germ cells by less than 10 rads (reference 6), the body can largely replace such cells and thereby repair the early somatic effects of sublethal doses. Hence this category of injuries endangers life only in the event of relatively high-level irradiation (several hundred rads absorbed within a few days).

Delayed somatic effects, on the other hand, represent a category of injuries which are comparatively irreparable and which may be caused by relatively small amounts of radiation absorbed in successive exposures over an interval of months or years. Consequently, these effects are of fundamental concern in the establishment of the maximum permissible dose. They include shortening of the life span, induction of leukemia and other cancers, and production of lens cataracts and degenerative changes of diverse types. Collectively, these effects resemble diseases occurring naturally in senescence and imply that irradiation hastens the onset of deleterious changes that would otherwise tend to occur spontaneously with advancing age. A distinction should be made, however, between effects such as cancer, which may seriously affect only a few of the individuals exposed, and life shortening, which may slightly affect all.

The earliest indication that irradiation might increase the probability of cancer was the observation, first reported in 1902, that skin cancer might arise as a complication of radiodermatitis.[7] Although many pioneer radiologists developed this disease through the practice of focusing their fluoroscopes on the bones of their own fingers and hands, skin cancer has long

since ceased to be an occupational disability among radiological workers because of improved protection standards.

Leukemia (cancer of the blood), however, is still abnormally prevalent in radiologists.[8] Furthermore, the incidence of this disease varies in relation to radiation dose in several other human populations, as well as in experimental animals, although the observed dose-incidence correlation has been restricted to relatively high dose levels (i.e., above 50 rads) in all instances to date.[9] Because of statistical limitations in the available data and ignorance about the radiobiological mechanism of leukemia induction, it is not possible confidently to extrapolate from high dose levels to the range of the maximum permissible dose level.* Attempts to make such extrapolations give widely varying estimates of the leukemia incidence, depending on the dose-response relation postulated (see Table I-3), and imply greater reliability of the existing data than is justified.

Table I-3 ESTIMATED INCIDENCE OF LEUKEMIA INDUCED BY LOW LEVELS OF RADIATION

Radiation	Dose rate assumed, rem/year	Duration of exposure, years	Incidence rate of radiogenic leukemia: Linear one-hit theory*	Incidence rate of radiogenic leukemia: Two-hit theory possible range†
Natural background	0.1	70	25,000‡	1 to 100‡
Maximum permissible occupational exposure	5	40	32§	0.1 to 10§

*E. B. Lewis, Leukemia and Ionizing Radiation, *Science*, 125: 965–972 (1957).

†See reference 10.

‡Per 50 million persons in 70 years, as compared with nearly 200,000 natural-causes cases.

§In a population of 4000 over a 40-year lifetime of occupational exposure, as compared with 15 natural-causes cases.

Other types of malignant growth have also been observed to result from irradiation in human beings, and it is clear from animal experimentation that nearly any form of cancer may be induced by irradiation under appropriate conditions. Unfortunately, however, dose-response data for other tumors are even more fragmentary than for leukemia.

It has been shown repeatedly in rodents that radiation reduces the life expectancy and that the extent of life shortening varies in relation to the total absorbed dose and the dose rate. As yet, however, comparable reduction of longevity in irradiated human beings has not been documented. The age-adjusted death rate among radiologists has been reported to be above normal, but the abnormality remains to be verified statistically.[11] Likewise, although the average age at death of radiologists is 5 years younger than that of other physicians, this difference is ascribable entirely to the younger average age of living radiologists. Paradoxically, at the lowest dose rates investigated to date in experimental animals (namely, 0.11 r/day), mice show reduced mortality until late in life, and their mean survival time often exceeds that of nonirradiated controls, despite an increased incidence of certain tumors.[11] In view of the foregoing, it is obviously impossible as yet to predict with confidence whether shortening of the life span occurs in human beings exposed in the range of the maximum permissible dose level.

Opacification of the lens of the eye has long been known to occur after intensive irradiation, but the development of cataracts in otherwise healthy cyclotron physicists came as a surprise when first noted slightly more than a decade ago. From observations on Japanese A-bomb survivors and experimental animals during the intervening years, it is evident that relatively small amounts of fast-neutron radiation (50 to 100 rads) can cause significant lens injury, whereas at least 200 to 600 rads of X rays is required for equivalent damage. Furthermore, it appears that neutrons are more cumulative in their effects on the lens than successive exposures to X rays or gamma rays. These differences between neutrons and X rays in relative biological effectiveness (RBE) and intensity dependence are still not entirely explained. They are consistent, however, with the occurrence of cataracts as an occupational disease among cyclotron workers but not among radiologists. The contrast is all the more significant in view of the relatively high average accumulated yearly dose (i.e., 100 r) that radiologists are estimated

*For a discussion of this subject, see Austin M. Brues, Critique of the Linear Theory of Carcinogenesis, *Science*, 128(3326): 693 (Sept. 26, 1958).

to have received in the absence of protective equipment.[12]

Genetic Effects of Radiation

From a wealth of experimental evidence, it has long been postulated that the unit of heredity, the gene, may be altered by the action of a single ionizing particle and that the frequency of gene mutation varies directly with the radiation dose in many types of cells. In germ-cell precursors, however, which are the cells of utmost genetic significance for man and other animals, the mutagenic effectiveness of gamma rays has recently been found to depend on radiation intensity,[13] from which it has been inferred that the classical one-hit interpretation of mutagenesis is not universally applicable. Other experimental data further suggest that the process of mutation may be reversible under certain conditions. Although these developments encourage the belief that the hazard of inheritable genetic changes may not be as great from low-level radiation as heretofore estimated, the data do not justify the assumption that a radiation threshold exists below which mutations are not induced.

The implications of genetic effects for human safety cannot be adequately assessed in the light of present knowledge. Nevertheless, since the overwhelming proportion of known mutations is deleterious in experimental organisms, as well as in human beings, it is assumed that any radiation-induced increase in the mutation rate above the natural level will be harmful. As yet, however, the magnitude of the damage that might result from a small increase in the mutation rate cannot be foreseen because the inheritance of most traits is poorly understood and complex; i.e., it is influenced by the interaction of multiple genes rather than by a single gene. It cannot be estimated, therefore, how much the intelligence, fertility, and vigor of a population would be affected, although changes in the incidence of certain simply inherited abnormalities might be tentatively predicted. Whereas some geneticists believe that man is already so heavily burdened with unfavorable genes that further impairment by deleterious mutations would endanger his future, this speculation is probably unduly pessimistic.[14]

All ionizing radiations are thought to have qualitatively similar biological effects, although they are known to vary in quantitative effectiveness. In general, the RBE of a radiation source varies within limits in relation to its linear energy transfer (LET). Hence, heavy particles have a high RBE for most effects when compared with gamma rays.[15] Because the RBE depends on the LET, it is influenced by dose rate; i.e., the biological effectiveness of high-LET radiation is less intensity-dependent than that of low-LET radiation, with the result that the RBE of the former increases relative to that of the latter with decreasing dose rate. For most biological effects investigated to date, RBE values for heavy particles range between 1.0 and 10. Consequently, it has been a policy to consider the relative toxicity of heavy particles for health physics purposes to be 10, in order to afford a reasonable margin of safety for personnel exposed to such particles under widely varying conditions.

Validity of Existing Protection Standards

From the foregoing, it is evident that existing data are not adequate to enable precise estimation of the biological hazards of low-level irradiation. Not only are there no unambiguous dose-response curves for any somatic or genetic radiation effects, particularly in the low dose range, but the influence of factors modifying the dose-response relation (i.e., dose rate, LET, volume of tissue irradiated, genetic constitution, age at irradiation, etc.) is poorly known. Whereas it was assumed in the past that small amounts of radiation could be tolerated without injury, experts are becoming increasingly reluctant to rely on the assumption that there is a threshold dose below which radiation is harmless. Instead, responsible authorities prefer to estimate the potential hazards of low-level irradiation, however uncertain such estimates may be, and to formulate protection standards by balancing potential risks against benefits.[2]

The existing standards[2,16,17] reflect an effort to limit exposure of the population, as well as of individuals who work with radiation, without unduly impeding the development of atomic energy and related fields. Although it is not assumed that the presently recommended maximum permissible dose levels are entirely without risk, these levels are expected to limit exposure to the degree that genetic and somatic effects will not exceed the bounds acceptable to society and commensurate with other hazards of modern civilization. (A. C. Upton)

A History of Nuclear Standards for Safety

The problems encountered in the nuclear field in the establishment of standards are more complex than those encountered in the establishment of standards for industry in general. The operational methods of industry have been developed gradually over the past 100 to 150 years, and, by the time the need for formalizing safety concepts was felt, much information on working practices applicable to occupational hazards was already available for sorting, organizing, and distributing to the interested parties. Such nonnuclear standards are developed in the United States by individual organizations and groups of organizations, including cooperating technical societies, industries, government agencies, and other nonbusiness organizations, and many of these standards are subsequently offered to international groups for consideration as international standards. Some industrial standards or codes have been incorporated in whole or in part in the laws and regulations of federal, state, and local governments. The development and activation of codes and standards have, in many cases, been based on common-sense principles that are understandable to all. For example, the dangers presented by improper electrical insulation or by substandard pressure vessels have been made obvious over the years through newspaper headlines about explosions and fires that caused loss of life and property. Strong public opinion arose against those who were willing to expose others to danger in order to cut expenses, and the enactment of protective laws began.[18]

The nuclear industry has developed in a different manner, and the development of standards for nuclear safety has taken a correspondingly different path.[19–21] When the restrictions were lifted and the public became aware of the new technology, nuclear energy was still under complete federal supervision, but, as the interest and involvement of private industry grew, rules and regulations for nongovernment operations had to be worked out. The original operating rules were developed under the stress of wartime urgency, and it is a great credit to those responsible that the safety problems were taken into consideration and were so successfully handled that accident casualties in the atomic energy plants were, and have remained, comparatively low. Detectable radiation injuries have been so few as to make them real items of news when they do occur.

The unexpected emergence of widespread public doubt (or public fear) became an important factor in the development of safety standards for the new industry. Although many hazards in other industries are serious, they are understood and accepted. Casualties from fire, electric shock, chemical and mechanical mishaps result from forces or causes detectable by the senses without instruments, and the damage versus intensity relations are understood; therefore the risks attendant to the benefits obtained are accepted. Since some dissenting opinions oppose the generally favorable claims for potential benefits versus risks from atomic energy utilization, more acceptable and convincing evidence of the real balance offered by atomic energy must be presented. The development of standards acceptable to the public by which risks can be measured for comparison with potential benefits thus has an important stimulus. One of the immediate aims of standards development has been the clarification of safety considerations for the public,[19,45,46] especially for the technical public.

The need for nuclear standards was made especially urgent by the fact that, with the relaxation of secrecy and with government policy encouraging the participation of private enterprise, a large body of relatively untrained persons began to come into contact with the various aspects of radioactivity. Since the first nonmilitary nuclear energy development work available to the technical public involved the use of radioisotope tracers in medical, biological, agricultural, and industrial fields, some of the first standards developed referred to the use of isotopes.

The purchase, transportation, and handling of radioisotopes are controlled by the AEC Division of Licensing and Regulation, which has also given attention to the detailed physical aspects of work with radioactive material and to the degree of competence of the individuals assigned to this duty.[22] A college education in traditional engineering or scientific disciplines is no longer a sufficient background for evaluating nuclear safety problems, because the new science has not yet become a part of the curricula in all colleges. Special training programs, such as those offered by the Oak Ridge Institute of Nuclear Studies, fill this need and can perhaps be called a basis for a minimum educational stand-

ard for the routine practitioner of the new science.[23]

The importance of the safety-associated specifications of working-area details and worker training requirements becomes apparent when the new dimension involved in the special character of the materials is taken into consideration. In addition to injuring himself, a worker in this field might expose not only his immediate neighbors but large populations to the possibility of radiation damage. One worker might, for example, accidentally release an excessive amount of radioactive materials into the atmosphere or into ground water, either of which could thus transport the danger over considerable distances. These considerations led to another set of specifications that set forth the maximum amounts of the various radioactive materials which can be tolerated in the atmosphere or in public waters. The generally accepted standards[17,24] specify maximum occupational exposures, exposures lower by a factor of 10 for occupants of the immediate vicinity, and, finally, a still lower maximum exposure for the general public. The actual values are based on the criterion that continuous exposure to the stated materials, including their ingestion, should not be significantly harmful to the population.

The establishment of these levels as maxima not to be exceeded is based on a vast, but still incomplete, body of data obtained by medical, biological, and health physics studies. The permissible body burdens were calculated from the experimental data on the basis of these studies. After careful consideration by national and international bodies,[17,24] these specifications have been tentatively accepted and are used as temporary criteria, while research continues to develop the knowledge and data required for further standards to be established.

Since the waste effluents of nuclear installations are potential sources of contamination, the various state and municipal governments involved take great interest in the proposed standards and use them to various degrees as bases for protective laws and ordinances. Some states have developed extensive programs of control and development, but other states have done very little in either field.[19–21] Legal considerations are frequently subjected to political pressures, and therefore a complicated set of contradictory standards has been issued by government agencies below the federal level. Thus a state (e.g., Pennsylvania) may have a legal limit of activity concentration of a certain material in a stream which is below that of the natural radioactivity and thus may indirectly forbid the disposal of any radioactive materials into streams. This problem can be solved temporarily by shipping the waste into another state for disposal, but this is obviously an undesirable solution, since prohibitive action can be taken at any time by the other state.

Most of the difficulties are caused by lack of knowledge and understanding, and uninformed political pressures could be neutralized by vigorous public education. Extensive analyses of atmospheric dispersion[25] and of the ecology of rivers[26] have been made, and a sound body of knowledge is available on which to base standards for disposal of radioactivity. The facts must be emphasized and disseminated in such volume and clarity that the general public can understand and judge them.

In the last analysis, standards represent a balance between the requirement of protecting the public, the benefits to be derived, and the necessity of disposing of radioactive wastes. Since the first requirement is paramount, operators of the various facilities have usually had to find means of storing waste material until the radioactivity has decayed to a satisfactory low level. In at least one case, a means was found for removing the radioactive material from the waste stream to a safe degree, while simultaneously obtaining useful production of radioisotopes for processing and sale.[27]

Concern that the maximum permissible concentrations of individual nuclides may be exceeded is based on the possibility of the accidental uncontrolled escape of radioactive material. Facility design must therefore be such that, even in the worst accident that could be reasonably expected to occur (the "maximum credible accident"), the amount that could be released would not result in contamination exceeding some as yet unstipulated maxima. This requirement leads immediately to escape-path considerations and thus to various standards involving pressure vessels and containment vessels. Most standards for these vessels are based on the *Boiler and Pressure Vessel Code*.[28] Strict compliance with these specifications greatly reduces the probability of an accident or, in case an accident does occur, of a harmful activity release.

Once activity is released, its presence and concentration must be established reliably. To

accomplish this, the work site, the waste streams, and the surrounding area must be monitored. A standard for monitoring methods and the instruments to be used has been developed by the National Bureau of Standards.[29] Decontamination procedures and standards for use in recommissioning a facility after an activity release have also been codified in a National Bureau of Standards Handbook.[30]

Handling of new or used fissionable materials presents the additional problem of possible criticality. Scientific considerations give a tool for evaluating this hazard, but it has required special efforts to reduce the physics involved to detailed engineering specifications. Considerable work has been done in this regard, and several guides dealing with the problem are available.[31–36] The chief hazard is still, however, the human element. For example, even in a plant designed for maximum safety by specifying allowable dimensions and allowable geometry of relative locations of the components, inadequate administration can allow a worker to bring into the operating area an oversized drum in which a critical mass can be formed. Recognition of this possibility emphasizes the importance of training, not only the technical supervisors, but all employees in all pertinent phases of nuclear safety. Organization of supervisory practices is an important phase of nuclear safety. Two examples have been described recently.[37,38]

The American Standards Association has organized a Nuclear Standards Board[39] that is developing standards covering all safety aspects of nuclear operations. Reactor safety, including questions which range from site evaluation to control requirements, is the subject of inquiry by Committee N-6.[40] This and other organizations producing standards directly involving safety in the use of nuclear energy were described in the March 1960 issue of *Nuclear Safety*.[19]

A glossary of terms has been published.[41] This aid is being expanded on an international level by the publication of multilingual glossaries by the United Nations, an edition of which was issued at the time of the Second Geneva Conference.[42] This is very important because, as nuclear science moves toward increased international collaboration, there is need for increased assurance that the terms used by scientists and engineers of one country can be understood without change in meaning by their counterparts in other countries. Thus standards for safety encompass the whole spectrum from the pressure-vessel code to the translation of terms like shim rod, scram button, coffin, and rabbit.

The importance of the field of standards is reflected by the rapidly growing body of literature on the subject. A recent literature survey[43] of standards reports and literature published since 1957 includes 296 references. Semicontinuous revision of standards is a recognized necessity, especially in nuclear energy adaptations, because of the rapid expansion of the industry and the concomitant potentials for higher living standards or for damage.

(Francois Kertesz)

JCAE Hearings on Radiation Protection

The Subcommittee* on Radiation of the Joint Committee on Atomic Energy (JCAE) held hearings on the basis and use of radiation protection criteria and standards on May 23–26 and June 1–2, 1960. In his opening remarks, Subcommittee Chairman Chet Holifield said:

> We hope these hearings will serve to clarify some of the confusion and misunderstanding by the public, as well as the scientific community, concerning the basis and use of radiation protection standards. I believe our print of Selected Materials[44] has provided the means of a great deal of clarification already.
>
> It is also hoped that these hearings will help us achieve a better understanding of organizational arrangements at the Federal and State levels of government through which radiation protection standards are developed and applied.
>
> Much discussion is needed also in the question of relationships between various private organizations, such as the National Committee on Radiation Protection and Measurements (NCRP) and the International Commission on Radiological Protection (ICRP), and the agencies of the Federal Government having important functions and duties in the radiation protection field.
>
> It is possible that the Joint Committee will want to consider, at a later date, ways in which Federal organization could be changed or modified, through legislation, if necessary, to permit a more effective process for the development and application of ra-

*The Subcommittee on Radiation of the Joint Committee on Atomic Energy consists of nine members from each house of Congress.

diation protection standards for use by Federal agencies and agency contractors.

A 1244-page preprint[44] of background information and statements had previously been distributed to the Committee. The preprint is a virtual encyclopedia of statements, reports, and information on the status of governmental actions, positions, and philosophies, as well as those of private industries, standards organizations, technical societies, medical and biology groups, labor organizations, and of individuals selected for their standing in fields significant to the subject. No attempt will be made here to summarize the data and opinions of the various persons and organizations which were presented to the Committee, and the reader is encouraged to refer to the Committee report for such information. The Federal agencies contributing to the hearings did not include the recently formed Federal Radiation Council. The 1960 report of the Council, which is parallel in interest to the hearings, was reviewed in the previous issue of *Nuclear Safety*.[45]

The essence of the AEC statement on radiation protection criteria is that, with minor exceptions, the Commission accepts and enforces the recommendations of the ICRP[17,46] and the NCRP[47] for all nonmilitary atomic energy activities under its jurisdiction. The implied philosophy of the Commission is to weigh all the known factors and then to set the criteria for enforcement toward conservatively safe practice. The Commission position regarding relations between hazards and exposure to radiation was previously presented in hearings before the JCAE[48–50] in 1959 and was not repeated in the 1960 statement. (L. A. Mann)

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Section II

ACCIDENT ANALYSIS

Burnout Heat Flux Correlations

The phenomenon of burnout is known to some extent to everyone who has been connected with nuclear reactor programs. The problems associated with burnout are extremely complex, and the results of burnout can be quite serious. The general aspects of burnout may be described in the following terms.

In the nucleate boiling regime, an increase in heat flux results in an increase in surface temperature if the fluid conditions remain the same. The additional heat flux that can be transferred per increment of wall temperature is much greater than that for ordinary convective transfer. Apparently the reason for this is that, as the bubbles leave the surface, they destroy the laminar boundary layer, which presents the greatest resistance to heat removal from the wall. As the heat flux and the wall temperature increase, a point is reached where the bubbles tend to coalesce and form a film. At first, this film does not completely blanket the surface but, rather, breaks away from the surface just as the bubbles did. The point at which the unstable film starts to form is referred to as the point of "departure from nucleate boiling" (DNB). Some authors refer to this point as burnout, and it is the flux for this point which is given by the Westinghouse correlations presented here. As the heat flux and surface temperature continue to rise, the film becomes more stable, until a point is reached where the entire surface is blanketed. In general, this point is referred to as burnout. If this burnout point is passed, the wall temperature will increase rapidly, even if the heat flux is reduced. The heat flux at burnout is usually less than 2 per cent above that at DNB and is never more than 5 per cent greater.[1,2]

The prediction of the burnout flux is based, in general, on conditions at the point of burnout. The effects of numerous parameters on the burnout flux have been studied, and a brief summary of these effects is presented below:

1. *Mass Rate of Flow.* The burnout heat flux is an extremely sensitive function of mass rate of flow or velocity.[3,4] The burnout flux increases with increased mass velocity at constant pressure and constant inlet subcooling for a given channel length-to-diameter (L/D) ratio.

2. *Subcooling.* The burnout heat flux increases with an increase in inlet subcooling or subcooling at the point of burnout.[3] An interaction between subcooling and the mass rate of flow has been reported.[4] The interaction is such that the effect of increasing the subcooling is higher at lower velocities.

3. *Pressure.* The effect of pressure on the burnout heat flux is somewhat more complicated than the effects of mass rate of flow and subcooling. With all other parameters held constant, the burnout flux increases as the pressure is reduced from the vicinity[3] of 2000 psia and increases as the pressure is increased from near atmospheric pressure.[4] Thus the effect of pressure is double valued. It is reported that, all other things being held constant, the burnout flux at 830 psia is equal to the burnout flux at 2000 psia and that at 1215 psia it is approximately 30 per cent higher than at the other two pressures.[5]

4. *Channel Dimensions.* It has been found that the burnout heat flux decreases with increasing L/D. Analyses have indicated that, for the same mass velocity, inlet subcooling, pressure, and L/D, the burnout heat flux for round tubes is as much as 45 per cent lower than for rectangular channels.[3] In a pool of water (no forced circulation) the burnout flux increases as the diameter of the heater in-

creases to a maximum flux at 0.06 in. and increases as the heater length decreases to a maximum flux[6] at 8 in.

5. *Orientation.* In forced-circulation systems, there appears to be no effect of orientation of the channel on the burnout flux.[3] It has been reported that, in a pool (no forced circulation), burnout of a solid vertical heater occurs at ~77 per cent of the flux for burnout of a horizontal heater.[6]

6. *Material.* The type of material and the surface finish, up to a roughness of 120 μin., have no effect on burnout flux in a forced-circulation system.[3,4] In pool boiling, there is some evidence that aluminum gives a higher burnout flux than other metals.[7]

7. *Gas Concentration.* Dissolved hydrogen in quantities of up to 135 cm^3 per kilogram of water at 2000 psia showed no effect in tubes 0.186 in. in inside diameter and 12 in. long.[1]

There have been numerous correlations proposed for computing the burnout heat flux which use the parameters discussed above. In general, they have been developed as attempts to fit experimental data and, as such, are not completely general. Table II-1 presents a list of the most widely used correlations (taken from references 4, 6, and 8 to 12), along with the test conditions for which they were developed. The symbols used in the correlations are defined as follows:

D_e = hydraulic diameter, ft
D_i = heated perimeter, ft
G = mass rate of flow, lb/(sq ft)(hr)
h_{BO} = heat-transfer coefficient at burnout point, Btu/(hr)(sq ft)(°F)
H_{BO} = enthalpy of fluid at burnout point, Btu/lb
L = distance from inlet to the burnout point, ft
P = pressure, psia
S = channel spacing, ft
T_B = bulk temperature at burnout point, °F
T_{wBO} = temperature of wall at burnout point, °F
V = velocity, ft/sec
ΔT_{sub} = subcooling, °F
ρ_f = density of fluid at saturation temperature, lb/cu ft
ρ_v = density of vapor at saturation temperature, lb/cu ft
ϕ_{BO} = burnout heat flux, Btu/(hr)(sq ft)

Each of these correlations should be used with care because of lack of complete generality. It should be noted that the Westinghouse correlations[3] use enthalpy at the point of burnout rather than subcooling. Such a parameter has the advantage that it can be used in the quality regime. Obviously, the burnout flux does not go to zero as the subcooling approaches zero.

There have been several statistical studies made of the parameters affecting burnout flux,[4,13] and it has been shown that less than 25 per cent of the total degrees of freedom accounts for over 90 per cent of the total variation in burnout flux.[4] In effect, nearly all the variation is a function of mass rate of flow and subcooling (or enthalpy).

It seems reasonable to ask with what degree of confidence one may use the correlations listed in Table II-1. In order to obtain a 99 per cent confidence level that burnout will not occur, Westinghouse[3] uses a design heat flux that is less than 65 per cent of the burnout heat flux predicted by their correlation.

The experiments from which the correlations were developed differ in two significant ways from the conditions which exist in a reactor. First, most calculations are made for the steady-state conditions, whereas accidents might normally be expected under transient conditions. Second, the experiments were performed with a uniform heat flux, which does not usually exist in a reactor. An additional reduction of the design heat flux is suggested to allow for the uncertainties of comparing the test case with the actual case,[1] and the combined reductions lead to a design heat flux that is 54 per cent of the predicted burnout heat flux. It should be pointed out that the safety factor on power is always less than it is on flux because an increase in power decreases the subcooling as it increases the flux.

Studies have been made of burnout following transient power conditions.[14] Indications are that once quasi-steady boiling has begun during an excursion for periods greater than 15 msec with large subcooling, behavior of the system at any instant is not appreciably different than it would be for steady boiling. The burnout flux for these conditions does not seem to be greatly influenced by the transient state of the power.

The correlations given in Table II-1 are, as mentioned above, designed for use with uniform heat generation. Care must be taken if the equations are to be used with nonuniform heat generation. References 2 and 3 give some de-

Table II-1 BURNOUT HEAT FLUX CORRELATIONS FOR FORCED-CIRCULATION SYSTEMS

Originator	Reference	Correlation	Test conditions: Pressure, psia	Sub-cooling, °F	Mass rate of flow, lb/(hr)(sq ft)	Velocity, ft/sec	Remarks
Mirshak, Durant, and Towell	8	$\phi_{BO} = 4.79 \times 10^5 (1 + 0.0365\,V)(1 + 0.00507\,\Delta T_{sub})(1 + 0.0131\,P)$	25–85	9–135		5–45	Annular and rectangular channels; hydraulic diameters ranged from 0.21 to 0.46 in.
McAdams et al.	9	$\phi_{BO} = (400 + 4.8\,\Delta T_{sub})\,V^{1/3} \times 10^3$	100				
Gunther	10	$\phi_{BO} = 7 \times 10^3\,V^{0.5}\,\Delta T_{sub}$	100	100		5–40	
Longo	4	$\phi_{BO} = 1.48 \times 10^{-5}\,P^{0.113}\,\Delta T_{sub}^{4.69}\,(G/10^3)^{1.22(2.38 - \ln \Delta T_{sub})}$	15–115	8–46	2.2×10^6 – 6.7×10^6		
Zenkevitch and Subbotin	11	$\phi_{BO} = 396 G^{0.5}\,\Delta T_{sub}^{0.33}\,\frac{\rho_f - \rho_v}{\rho_f}$	2000–3000	50–200	0.6×10^6 – 3.6×10^6		
Bernath	6	$\phi_{BO} = \left[1.089 \times 10^4 \left(\frac{D_e}{D_e + D_i}\right) + \frac{48V}{D_e^{0.6}}\right] \times \left[103 \ln P - 97.2\left(\frac{P}{P + 15}\right) - 45V + 32 - T_B\right]$	500–3000			4.5–54	This is a composite of three formulas; Bernath[6] assumed that $\phi_{BO} = h_{BO}\,(T_{nBO} - T_B)$ and then proposed correlations for $(T_{nBO} - T_B)$ and for h_{BO}; the correlations were devised from data for round tubes
Jens and Lottes	12	$\phi_{BO} = K\left(\frac{G}{10^6}\right)^n \Delta T_{sub}^{0.22} \times 10^6$	500–2000	1–163	0.94×10^6 – 7.8×10^6		Condition I,* $L/D = 109$
			1000–3000	1–149	0.9×10^6 – 7.5×10^6		Condition II,* $L/D = 22$
DeBortoli et al.	3	$\phi_{BO} = 0.28 \times 10^6 \left(\frac{H_{BO}}{10^3}\right)^{-2.5} \left(1 + \frac{G}{10^7}\right)^2 e^{-0.0012\,L/D_e}$	1850–2150		0.2×10^6 – 8.0×10^6		Round tube
		$\phi_{BO} = 0.37 \times 10^6 \left(\frac{H_{BO}}{10^3}\right)^{-2.5} \left(1 + \frac{G}{10^7}\right)^2 e^{-0.0012\,L/S}$	1850–2150		1.6×10^6 – 5.0×10^6		Rectangular channel
		$\phi_{BO} = 0.50 \times 10^6 \left(\frac{H_{BO}}{10^3}\right)^{-2.5} e^{-0.0012\,L/S}$	1850–2150		0.2×10^6 – 5.0×10^6		Rectangular channel

* The constants K and n for conditions I and II as a function of pressure are:

Pressure	Condition I K	Condition I n	Condition II K	Condition II n
500	0.817	0.160		
1000	0.626	0.275	0.915	0.275
2000	0.445	0.500	0.545	0.500
3000			0.300	0.725

tailed suggestions for making computations with nonuniform heat generation. Experiments have been made with "hot patch" tests,[4] which indicate that flux at the hot spot may be as high as four times the flux in the rest of the section before burnout occurs. However, most tests showed a factor of between 1.0 and 1.5.

In conclusion, it should be pointed out that any summary of burnout heat flux becomes obsolete during the writing. However, the correlations presented indicate the importance of the various parameters, some of the difficulties to be met when dealing with the subject, and several approaches to its solution.

(J. F. Wett, Jr.)

Stored Energy in Graphite

The problems associated with stored energy in the graphite of both graphite-moderated reactors and reactors that employ graphite as a reflector material are being studied. In both types of reactor, the major concern lies with the possible effects of accidental release of a large quantity of thermal energy. In large graphite-moderated reactors, there is the possibility that the fuel elements might be subjected to very high temperatures as a result of such an energy release. In reactors employing graphite as a reflector, on the other hand, attention must be given to the effects of stored-energy release on the reactor structure. The fuel elements are not normally endangered in this case, because they are usually situated in a separate containment shell with a separate cooling system.

Comparison of BEPO and ORNL Annealing Methods

The British graphite-moderated, air-cooled reactor BEPO was annealed in early 1958 in a planned release[15] employing externally heated air blown into the normal coolant duct of the reactor while the reactor power was off. In preparation for that release, theoretical studies[16] were conducted to provide guides to heatup rates based on a constant-activation-energy model. Since the reactor must be annealed again, the validity of the theoretical model used previously has been reexamined.[17] Although there are no discrete activation energies describing the release of energy stored in graphite, Rimmer[17] makes use of the fact that the spectral distribution of activation energy as a function of stored energy is relatively flat. Moreover, he weighs his choice of activation energy more heavily toward the lower stored energy portion of the curve, since it is in this region that an actual in-pile release is most important.

In order to test the validity of the constant-activation-energy model, samples of Windscale Pile graphite were annealed by a quasi-adiabatic rise method in an apparatus that simulated coupling of the graphite, with a time constant of 10 min, to air heated at various rates. At low rates of rise of air temperature, the results showed that the graphite temperature closely followed the air temperature; whereas at higher rates the results showed that the graphite temperature deviated from the air temperature, and a quasi-adiabatic rise in the graphite temperature was achieved. The boundary between these two types of behavior is sharp, and Rimmer used his model successfully to predict the air heating rate at which the changeover would occur. It was found that the critical rate was close to 1°C per minute and that the quasi-adiabatic condition was achieved at rates of 2 and 4°C per minute. The activation energies selected for these predictions were 1.2, 1.4, and 1.7 ev, and all appeared to work satisfactorily.

This type of analysis is excellent for the preparation of an annealing schedule for BEPO, not only because it provides the basis for the operational equipment and timing, but also because it provides the necessary information for a "successful anneal." The term "successful anneal" in this sense is meant to be a maximum rise in the graphite temperature that is below any dangerous temperature for any of the reactor components. This approach to annealing graphite is quite different from that used for annealing the ORNL Graphite Reactor, as described in the December 1959 issue of *Nuclear Safety*,[18] and it is interesting to compare the two approaches, since the reactors are very similar.

A "successful anneal" using the reversed air-flow technique[19] employed at ORNL first removes the dangerous portion of the stored-energy-release spectrum that can be spontaneously triggered by employing low heating rates from fission power. Higher annealing temperatures are then used when no thermal excursions

are possible. The ultimate objective is, of course, to heat the graphite to a high, yet safe, temperature, but the basic difference between the two schemes is that the BEPO approach utilizes the stored-energy release itself as a source of power to reach the desired temperature, whereas the ORNL approach masks the stored-energy release and achieves the desired maximum annealing temperature by close control of the fission heat.

Stored Energy in Reflector Graphite

The stored energy in the NRX reactor reflector graphite at Chalk River, Ontario, was measured[20] in 1958, and the results indicated that there was no present danger of accidental release of the energy stored in the graphite reflector. Of considerable interest was the fact that the energy stored in the reflector in 1958 was smaller by a factor of 5 to 10 than the stored energy found in 1953, although no controlled stored-energy releases were undertaken between 1953 and 1958. Following the boost in reactor power from 30 Mw to 40 Mw in 1954, there appears to have been continuous annealing because of the combined effects of radiation annealing,[21] first reported by Hanford workers, and thermal annealing from the increase in graphite temperatures accompanying the rise in reactor power. In addition, a reexamination of graphite temperature records revealed that thermal release took place on July 1, 2, and 3, 1955, when temperatures climbed as high as 35°C above normal. The temperature records indicated no other stored-energy releases of comparable magnitude. All the stored-energy measurements showed that the maximum stored-energy release rates never exceeded 0.08 cal/(g)(°C), and, if this is representative of the most serious situation that exists, it is clear that no danger exists. It is pointed out, however, that the temperature distribution is not well monitored in the reflector graphite and that it would be desirable to have more thermocouples placed at strategic positions. Additional graphite stored-energy measurements were to be made in the summer of 1960 so that stored-energy growth rates could be determined.

Unlike the situation with respect to the NRX reflector graphite, a proposal[22] has been made to anneal the stored energy in the MTR reflector graphite while the stored-energy content is low and easily controllable. The proposal indicates that, at present power levels, the energy accumulated *during* the next five-year period cannot cause an excessive transient structure temperature if an accidental release of that energy occurred. It is suggested that the stored energy be released by thermal annealing with fission power as a source of heat at low reactor power and low air flow.

The proposal is based on a comprehensive heat-transfer analysis of the effect of a thermal transient resulting from a stored-energy release in the pebble-bed zone of the graphite reflector. For the calculations the worst credible conditions were assumed, including failure of the cooling fans, no stack draft, a reactor scram from the 40-Mw equilibrium condition, and instantaneous release of the total predicted stored energy of the pebble bed. These conditions were employed to determine the stored-energy condition as it existed in 1959 and for extrapolated times of five years and infinity. In substance, the calculations indicated that no dangerous condition would develop for at least five years but that prolonged operation beyond that time might result in a more hazardous situation. In addition, a heat-transfer analysis based on the proposed annealing scheme indicated that the maximum peak temperature of the annealing transient would not exceed 260°C and would, therefore, present no danger so far as graphite combustion is concerned.

A complete preannealing, annealing, and post-annealing schedule has been outlined in detail. In particular, the annealing scheme calls for reduced air flow operation, together with stepped increases in power after 30-min intervals, with the maximum rate of temperature rise being held to 400°F per hour before a stored-energy release is initiated at about 300°F. At 350°F the power is to be reduced 50 per cent, and at 500°F the reactor is to be scrammed by automatic action. The fans are not scheduled to be turned on automatically unless a temperature of 600°F is reached.

These analyses of the stored-energy problem in the graphite reflectors of the NRX[20] and MTR[22] reactors leave some interesting questions unanswered. Since the two reactors have operated at comparable power levels (40 Mw) for similar periods of time with what appear to be not too different reflector temperatures, it would be expected that rather similar graphite stored-energy conditions would have developed. For this reason it is not completely clear why

thermal and radiation annealing should have occurred in the NRX reactor[20] in a very pronounced manner, while only an apparently minor thermal anneal has occurred in the MTR reactor, as revealed by a slight shift in the peak damage position.[22] It is conceivable that large differences in the fast-flux spectra in the two reactor reflectors might contribute to these apparent discrepancies. Perhaps even more important is the possibility of larger temperature differences in the reactor than are presently known. (M. C. Wittels)

A Summary Description of the SPERT Experimental Program

By J. C. HAIRE, Jr.*

The ultimate goal of the AEC program of research and development in reactor safety[23] is to provide the factual data upon which satisfactory evaluations of reactor safety can be made. As a major part of this program, experiments are being carried out in the SPERT (Special Power Excursion Reactor Tests) facilities. The objectives of the SPERT program have been discussed in detail by Nyer and Forbes.[24,25] The primary aims may be summarized as being the development of information on the kinetics of reactor systems and the inherent physical mechanisms which affect the neutronic behavior and thus the safety of the reactor. The program pursued up to the present time has dealt almost exclusively with investigations related to reactor runaway excursions in heterogeneous water systems, but it is not intended that the program be limited to studies of those systems. In fact, much of the information already obtained can be applied to reactors other than heterogeneous water systems.

A number of available documents contain descriptions of the SPERT work.[24–57] For example, a recently published report by Nyer and Forbes[25] briefly outlines the history of the program, cites reviews and proposals which, although not related per se to the reactor-system kinetics aspects of nuclear safety, aid in establishing the framework upon which considerations of future program direction must be based, and discusses in detail the experimental and analytical accomplishments of the program through 1958. The considerable amount of information that has been gathered since the report of Nyer and Forbes was prepared is documented primarily in quarterly reports.[37–39] Some of the current efforts are described in summaries of papers presented at the December 1960 winter meeting of the American Nuclear Society.[53–57]

In this review the aim is to indicate the various paths of investigation followed and to summarize significant results.

Facilities

Concise, but complete, descriptions of the SPERT-I, -II, and -III facilities are presented in a summary report[44] of design data which also includes tabulations of some operating and physics data. No detailed description of the design of SPERT-IV has yet been published, but some data are available.[36]

The major design characteristics of the reactors are summarized in Table II-2, and indications of the nature of the kinetics problems being investigated in each of the facilities are given.

The results of many of the experiments performed on each of the facilities have a general applicability and are not necessarily restricted to the particular type of system which each facility approximates as a prototype by virtue of its system parameters. The features and experimental program of each reactor are described briefly below.

SPERT-I. The purpose of SPERT-I is to provide a facility for conducting basic transient tests and safety studies. The reactor vessel is open to the atmosphere and there is no provision

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Table II-2 SPERT FACILITY DESIGN CHARACTERISTICS[36,44]

	SPERT-I	SPERT-II	SPERT-III	SPERT-IV
Moderator	H_2O	H_2O or D_2O	H_2O	H_2O
Reflector	H_2O	H_2O, D_2O, or solid	H_2O	H_2O
Vessel inside diameter	4 ft	10 ft	4 ft	20 ft (each section)
Vessel height	15 ft	16 ft	19 ft	25 ft
Max. system pressure	Atmospheric	375 psi	2500 psi	Atmospheric
Max. moderator temp.	212°F	400°F	670°F	212°F
Max. coolant flow rate	0	20,000 gal/min	20,000 gal/min	5000 gal/min
Coolant heat-removal capacity	None	None	60 Mw for 30 min	1 Mw

for heat removal. Experiments are limited to operation at atmospheric pressure with initial bulk water temperatures not in excess of boiling. Coolant flow velocities are limited to those attainable from convection; however, in one series of tests[34] a mechanical stirrer and flow-baffle arrangement were used to provide forced coolant flow up through the core with estimated velocities of a few tenths of a foot per second. The comprehensive kinetic information available from such a water-moderated and -reflected system provides basic data for analysis of reactor kinetics and design.

The reactor went critical in July 1955, and transient tests were first performed in September of the same year. Since that time experiments have been conducted on six different cores of highly enriched, flat-plate fuel assemblies.[39,45] Four of these cores were aluminum clad and two were clad with stainless steel. The six cores provided a considerable variation in core parameters. Both step and ramp reactivity insertions were investigated over a wide range of reactor periods and reactivity-insertion rates. The effects of the void and temperature coefficients of reactivity, initial temperature, initial power, hydrodynamic head, moderator surface tension, and dissolved gas in the moderator have also been studied.[45,46]

Although studies of the reactor are restricted by the design parameters indicated in Table II-2, the capabilities of the reactor can be extended and a wide range of experimental parameters and conditions can be achieved by using the reactor as a driving source for subassembly tests. Some tests have been performed to investigate the details of specific shutdown mechanisms under pressure conditions approaching those available in SPERT-III.[37]

Tests are scheduled to start in January 1961 in which a study will be made of the effects of a long thermal time constant in the fuel, the kinetics of a core with low fuel enrichment (~4 per cent), and the shutdown effects of prompt coefficients that do not depend on heat transfer. In addition to the different nuclear properties obtainable with this core, which is composed of fuel pins from the N.S. *Savannah* critical facility,[39] it is expected that a hydrodynamic situation to be provided in the reactor will be different from anything previously encountered in a SPERT facility.

SPERT-II. The purpose of SPERT-II is the study of the effects on kinetic behavior of the neutronic properties of a reactor, as determined by the moderator and reflector. The system has provisions for the use of H_2O, D_2O, or solids such as beryllium or graphite as the reflector material. It is also worth noting that this system could be converted fairly readily to use an organic material as the moderator. The coolant flow system provides for both upflow and downflow through the core.

The reactor went critical with an H_2O moderator-reflector in March 1960 and with a D_2O reflector in August 1960. Measurements have been made of static parameters for various critical core configurations in D_2O and H_2O. The experiments utilized close-packed lattice cores and spread-out lattices more typical of D_2O systems. Work related to the problems of low-level startup has also been carried out by investigating intrinsic neutron source levels in enriched uranium-aluminum cores.[56]

The immediate program of investigations for SPERT-II will use plate type cores in D_2O and H_2O. The initial aim is to determine the effect of changes in the prompt-neutron lifetime on kinetic behavior over the range of system parameters available in SPERT-II.

Safety evaluations related to problems of accident initiation in D_2O systems should be aided by the data obtained from the D_2O transient test program in SPERT-II.

SPERT-III. The purpose of SPERT-III is the investigation of reactor kinetics over a wide range of system temperature, pressure, and flow. The environmental conditions in the reactor make it suitable for investigation of the problems pertinent to pressurized- and boiling-water power reactors. Since December 1958, critical studies have been carried out[47] over the full range of design parameters shown in Table II-2. Both static and transient tests have been conducted for studying the separate and combined effects of initial system pressure and coolant flow on the system behavior at room temperature.[54,55]

A stainless-steel-clad, plate type core of highly enriched UO_2 fuel in a stainless-steel matrix is being used in SPERT-III for the initial test program which, in addition to the aforementioned tests, will also include studies of system temperature effects, excursions from operating power, loss of flow, and cold-water accidents. The loss of flow and the cold-water accidents are of extreme importance in hazards evaluations of power reactors.

SPERT-IV. The observation of extremely violent self-induced power oscillations in SPERT-I led[27,42] to a broadening of the SPERT program because it became clear that the stability studies originally scheduled as a part of the SPERT-I program required the hydrodynamics of a large pool for more extensive investigation. The SPERT-IV facility, which includes a large pool in two sections, is now being constructed to fill this need, to provide a prototype for safety tests of swimming-pool type reactors,[24] and to provide a facility for testing a variety of core types and sizes. In particular, the stability characteristics and self-limiting properties of large cores can be examined. The estimated completion date for the facility is July 1961.

Accomplishments

A consideration of the technical accomplishments in terms of applicability of the results provides an informative basis for a review of the SPERT program. The results of the kinetics investigations range in applicability from the most general, applying to almost all reactors, to the very particular at the level of empirical descriptions applicable only to a specific design. A major part of the analytical effort has been devoted to considerations of the general kinetic properties of reactor systems, and the results obtained apply to reactors of all types having reactivity-to-power feedback described by the very general form of the coupling equations assumed in the analysis. Of more limited application are the results of some of the experimental investigations of shutdown effects which, in general, can be expected only in the subclass of plate type water-moderated reactors. At the level of prototype applicability are some of the observed behavior properties of the SPERT-I reactor which would be expected only in systems of similar design. One further type of result is obtained from studies of specific physical phenomena that are known or expected to play a part in the neutronic behavior of reactors. These results have broad applicability because they constitute basic information on the mechanism of neutronic behavior.

The SPERT experimental and analytical accomplishments can be arranged into five major groupings according to the type of experiment involved. The five types of experiment are static experiments, tests with step reactivity insertions, tests with ramp reactivity insertions, stability tests, and tests of specific shutdown mechanisms. Within each of these categories, the full range of environmental conditions available in the facilities provides the opportunity to experimentally determine the effect on kinetic behavior of variations of individual system parameters.

Concurrent with the experimental work, there has been an effort to develop a theoretical framework for the prediction of reactor dynamic behavior under a given set of conditions from a knowledge of the system design. The influence of certain system properties on dynamic behavior has been sufficiently explicit in the analysis to permit a quantitative evaluation of the effects of various design modifications on the safety and operability of the reactor.[48,51] The latter point has in large measure determined the analytical approach taken in studying reactor kinetics problems.

Static Experiments. In the static tests, certain core characteristics, such as the void and temperature coefficients of reactivity, are measured to provide a basis for relating dynamic behavior to static parameters. This is an essential part of the experimental work which must be performed on each core in order to be

able to interpret the results of investigations of shutdown behavior.

Step Tests. In tests of the effects of step insertion of reactivity, the reactor is suddenly made supercritical and then the system behavior is observed by measurement of parameters which could be labeled damage indicators, i.e., power, temperature, pressure, and energy. This type of test simulates the most extreme accident from the viewpoint of accident-initiation rate. The information obtained from such tests aids in determining the relations existing between reactor behavior, reactor characteristics, and the kinetics equations. The relations thus identified describe the general properties of reactor excursions. The test results help to establish the relative importance of certain system parameters and indicate some of the specific physical mechanisms that determine the system behavior.

Many general characteristics of self-limiting power bursts in reactor systems have been revealed experimentally and predicted analytically. Foremost of these are the functional relations that hold at the time of peak power between the initial asymptotic reactor period, the reactor power, the energy release, and the reactivity compensated by shutdown mechanisms. The initial tests (references 26, 29, 40, and 50) conducted in SPERT-I revealed experimentally those features which had been predicted from earlier analytical work.[43] From a rather simple extension[36] of the early analysis, these relations were shown to be primarily a consequence of reactor kinetics equations. It was therefore expected that the principal features would be exhibited to some degree by all types of reactors. The expectation has been sustained by later experiments with other cores in SPERT-I and SPERT-III. These general characteristics were also exhibited by the BORAX, KEWB, TRIGA, and Godiva reactors.[58] It has been shown analytically that these features are related to the shape of the power burst produced by a given reactor and also to its characteristic neutron lifetime. Although this relation yields no information on the physical mechanisms affecting the reactor behavior, it is useful because the power-burst shapes used in the analysis include many of those which will be met in actual practice.

The effect of burst shape on the compensated reactivity relation was demonstrated experimentally in the SPERT-I tests initiated at boiling temperatures which produced sharper power-burst shapes than those obtained in the subcooled tests.[35] The experimental work in SPERT-II with D_2O in the system will provide an opportunity to check the predictions regarding the effect of prompt-neutron lifetime.

The shape of individual power bursts revealed a need for adjustment in the theory to provide a nonlinear relation between the energy release and the reactivity compensated. Incorporation of this dependence into the analysis led to a prediction that the reactor behavior would have a weak dependence on the shutdown coefficient, a quantity proportional to the ratio of void coefficient and prompt-neutron lifetime.[32] Variations in the shutdown coefficient of a factor of 50 were achieved in the cores tested in SPERT-I, and the weak dependence predicted by the analysis was verified.[45]

Some of the other results of the step transient tests are summarized below. The degree of applicability varies from the rather general statement of item 1 to the extreme limitation of item 10, which applies only to the system for which the result was obtained:

1. For transients up to and somewhat beyond prompt critical, the reactivity change required to halt the power rise was appreciably less than that originally injected.[29,41]

2. The reactor behavior was very reproducible, and extrapolation of results from longer periods to shorter periods could be performed without undue hazard.[40]

3. Nuclear excursions became less severe as the initial reactor temperature was increased toward the saturation (boiling) point.[34]

4. Increasing the hydrodynamic head by raising the depth of water over the core produced no measurable effect in the subcooled tests but increased the peak power and energy released in the boiling tests.[46]

5. The effects of forced moderator flow through the core of a few tenths of a foot per second in SPERT-I (reference 34) and up to 18 ft/sec in SPERT-III (reference 54) were an increase in equilibrium power and a broadening of the postpeak side of the power burst, even to the point of completely washing out the initial power maximum.

6. The effect of increasing the system pressure from 0 to 2500 psi was a slight increase

in the peak power and a broadening of the power burst which resulted in an energy release, at the time of the power peak, that was a factor of 2 greater than the energy release in the unpressurized tests.[55]

7. Increases in the starting power in the range of 5 watts to 100 kw produced no change in the maximum power attained.[46]

8. Modification of miscellaneous factors, such as the amount of dissolved gas in the moderator or the water surface tension, produced no measurable changes in power or temperature behavior up to the peak of the first power burst, although some postpeak changes were observed.[46]

9. The general applicability of the detailed test results to reactors of similar design was demonstrated by the essentially identical behavior of the SPERT and BORAX reactors.[40]

10. The degree of protection attainable from inherent mechanisms under extreme accident conditions in SPERT-I is as great as that afforded by the control system, although the control system may give additional protection in the case of limited reactor accidents.[53]

Ramp Tests. In the ramp reactivity-insertion tests, reactivity is added to the system at a constant rate to simulate the reactor accident considered to be most plausible of the types normally postulated. These tests differ from the step tests only in the type of accident simulated.

The general characteristics of step-induced power bursts were observed to apply also to ramp-induced power excursions.[40] The correlation between the results of step and ramp tests is made on the basis of an effective reactor period, which, for step tests, is the measured initial asymptotic reactor period and is determined by the amount of the reactivity insertion. The effective period for ramp tests is the minimum measured period, since, during a ramp-induced excursion, the instantaneous period shortens as the burst develops, passes through a minimum, and then increases. This equivalence of step and ramp bursts was also established in subsequent analytical work, which yielded, in addition, a means of predicting the minimum reactor period from the ramp rate.[32,45] Experiments have verified the predictions of the analysis over a fairly wide range of starting powers.[45] The results have general applicability if two limitations are met by the initial power: (1) it must be several decades below the level at which shutdown effects become manifest, and (2) it must be higher than the natural source level to ensure a deterministic ramp condition at the initiation of the reactivity insertion. In recent step tests at SPERT[53] with no artificial source in the core, delay times as long as 2 sec existed before a sustaining chain was initiated. Thus the second limitation on initial power is not a trivial one, since a time delay in the initiation of a sustaining chain reaction could produce, even for modest ramp rates, a disastrously large reactivity insertion before the reactor responded.

The number of ramp tests performed at SPERT has been only a small fraction of the total number of step tests performed, and they were not performed on all cores tested. Within these limitations, however, the results verify the theory and may be summarized as follows:

1. The most important parameter is the reactivity insertion rate, since it affects the transient behavior more strongly than either the initial power or the shutdown coefficient.[45]

2. Ramp and step excursions of the same effective period are essentially equivalent.

3. The simple model employed to describe steps, when properly modified, describes ramp tests and provides an analytical method for obtaining minimum period. Consequently, ramp behavior can be predicted from the results of step experiments, and step behavior can be predicted from the results of ramp tests.

Since all these results are primarily a consequence of the coupling postulated between reactivity and power history, they have general applicability to all types of reactor.

Stability Tests. Both self-induced and externally excited power oscillations provide information on reactor stability as a function of system conditions. The tests simulate accidents that could occur as a consequence of exceeding certain levels of control variables in an operating reactor. The test results provide information on the consequences of such an accident and help determine what means can be taken in an operating reactor to provide an indication of the proximity of this type of accident.

Some aspects of reactor instability have been investigated[27,42] in SPERT-I. In the tests the reactivity was added to the system either continuously or incrementally, and the reactor power behavior following decay of the initial

transients was observed. In some cases the violent power oscillations observed grew rapidly to destructive proportions and necessitated scramming the reactor to prevent core damage. The investigations were limited by the hydrodynamics of SPERT-I and will be extended in SPERT-IV; however, the following results are of significance in reactor stability considerations:

1. The rate and extent of void collapse observed during the oscillations were sufficient to permit successive short-period (~ 5 msec) transients to occur and be self-limited at the rate of one or two per second.

2. The tendency toward oscillatory behavior increased with the amount of reactivity addition above zero-power critical and also increased with the hydrodynamic head or depth of water above the core.

3. Reproducibility was not good, but it was established that a fine line existed between stable and unstable reactor behavior whenever the reactivity held in the moderator voids was of the order of 1.5 per cent Δk.

4. Significant shifts that were observed in the amplitude of the power oscillations could not be attributed to any apparent cause. These shifts occurred after several seconds of fairly steady oscillations of about 200-Mw amplitude, at which time the amplitude of the bursts would suddenly increase an order of magnitude, to about 2000 Mw, for no apparent reason.

In addition to the investigations of self-induced reactor oscillations, as described above, externally excited oscillations have been investigated by means of a pile-oscillator technique.[52] The oscillator work has provided a determination of the phase and magnitude of the reactivity-to-power transfer function from 0.01 to 18.4 cycles/sec at low power and at temperatures below boiling. Work at higher power levels has revealed significant changes in the transfer function at power levels above 300 kw. The principal objective of this part of the SPERT program is to determine to what extent pile-oscillator tests may be utilized to predict the dynamic behavior of a reactor in accident situations. It is to this end, i.e., complementing the program of transient testing, that investigation of the linear and nonlinear stability of the reactor systems by the pile-oscillator techniques has been undertaken.

The pile-oscillator technique also provides a means of investigating kinetic parameters, such as l/β_{eff}, the ratio of the prompt-neutron lifetime to the effective delayed-neutron fraction. A recent determination of l/β_{eff} from low-power oscillator data is in good agreement with the results obtained from analysis of transient data and with the measurement made by the poison-perturbation method.[57]

Shutdown Mechanism Tests. The investigation of individual physical mechanisms both in-pile and out-of-pile provides information on the details of processes affecting the neutronic behavior of the reactor and thus constitutes another segment of the program of kinetic investigations. This detailed information aids in establishing the system conditions under which a given mechanism may be effective, helps to determine the extent to which reactor behavior may be affected by component or design changes important to a given mechanism, and is essential to the successful extrapolation of some of the results obtained in SPERT type reactors to reactors of other types in which a common mechanism is known to occur.

In the case of the SPERT type reactors, the primary shutdown mechanisms are regarded to be those which are responsible for decreasing the effective moderator density.[45] This density coefficient of reactivity has contributions from thermal expansion of both the water and fuel plates, from steam void formation, and possibly from radiolytic gas production. One of the primary problems to be answered in this part of the work was to account for the way steam formation affected reactor shutdown.

Some experiments have been performed for the specific purpose of investigating the moderator-expulsion effects responsible for the self-limitation of the transient excursions. One set of experiments was performed in which the SPERT-IA core was coated with several layers of thermosetting plastic[36] in an effort to alter the heat-transfer characteristics of the fuel plates and thus to effect an isolation between the processes dependent on heat transfer, such as boiling, and those which are independent of heat transfer, such as radiolytic gas formation. If boiling were the chief contributor to initial shutdown, the time required for nuclear heat to pass through the plastic layer should affect the power levels ultimately reached in the transient bursts. The experimental results led to ambiguous interpretations in that they indicated that either boiling was not the chief contributor or

the plastic coating had not achieved the calculated thermal insulation of the plates. Subsequent experiments in closed fueled and unfueled capsules gave results contrary to the first interpretation;[36] that is, these tests indicated that heat transfer was an essential element in self-shutdown and that no appreciable shutdown effect could be attributed to processes not involving heat transfer. The results of the capsule experiments appear to be unambiguous. More detailed experiments of the capsule type[37] demonstrated that pressure and volume changes occurred simultaneously with a rise of plate temperature through the boiling point. Within experimental error, no delays were observed, regardless of initial period or pressure.

In-pile photographic investigations of boiling[37] within a fuel-bearing capsule have provided documentation that the magnitude of bubbles around the fuel plates increases so rapidly following the initial appearance of nucleate boiling that boiling does contribute significantly in the shutdown of reactors when saturation temperatures are reached by the fuel-plate surfaces. The results of the capsule experiments constitute strong evidence for boiling as a principal shutdown mechanism for atmospheric tests and thus essentially complete the case for disregarding inferences from the insulated core experiment.

In an effort to determine analytically the relative contributions to reactor shutdown of the known effects, the reactivity changes resulting from moderator heating by conduction from the fuel plates and from expansion of the fuel plates have been calculated from the observed power behavior for different values of the reactor period. An analytical solution to the thermal-diffusion equation, using a two-term approximation for the power burst, permits a calculation of the distribution of heat energy and the resulting reactivity changes. These calculated reactivity changes at the time of the power peak have been compared with the experimentally determined values to see how well straightforward calculations would account for the observed changes in reactivity. The calculations included the effects of nonlinear expansion of the moderator, neutron and gamma heating of the moderator, and nonuniform distribution of density changes in the core. The results may be summarized as follows:

1. For long-period transients ($\tau > 200$ msec), calculated effects are smaller than observed reactivity changes. Conduction heating, assumed to be the predominant heat-transfer mechanism, in causing a moderator density change accounts for most of the reactivity change; the deficit may be provided by radiolytic gas formation.

2. In the intermediate-period region (40 msec $< \tau <$ 200 msec), fuel-plate expansion, which in turn causes some moderator to be expelled, accounts for all the observed reactivity changes.

3. In the short-period region ($\tau <$ 40 msec), the calculated effects fall short of accounting for the observed changes; however, the inclusion of a steam term whose shape is calculable but whose magnitude must be fitted yields a good fit to experimental data.

One additional bit of information available[55] from SPERT-III concerns boiling as a contributor to shutdown. In tests of the effect of raising the system pressure for some ambient step-transient tests, it was found that, by increasing the pressure to 2500 psi, boiling could be eliminated as a shutdown mechanism for periods as short as 11 msec.

Summary

These concurrent experimental and analytical investigations of kinetic behavior have established the general behavior characteristics of reactors of this type and have permitted some predictions to be made for other types of systems. Mechanisms responsible for the behavior characteristics have been studied, and such studies will be continued.

Correlative models have been successfully employed to describe the experimental observations and to predict certain unexplored parametric effects. The use of correlative models provides an understanding of the kinetic behavior of a reactor system only in so far as it provides a method, based on some experimental results, for predicting what a system will do either as a consequence of changes in a parameter specified in the model or by identifying some physical phenomenon with a parameter specified in the model. Full understanding of kinetic behavior requires detailed knowledge of shutdown mechanisms based on physical principles. Although correlative models, successfully applied, do not permit a full evaluation of reactor-system safety without benefit of experiment, they do serve as a tool which could

enable the user to achieve gains in the inherent safety of a system on the basis of a minimum number of safety experiments which, in some cases, might consist in measurements of only static parameters. (J. C. Haire, Jr.)

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Section III

INSTRUMENTATION AND CONTROLS

Nuclear Process Instrumentation and Controls

At a conference on Nuclear Process Instrumentation and Controls,[1] sponsored by the ORNL Instrumentation and Controls Division and the AEC, papers on several subjects of major importance to safety in atomic-power generation were presented and discussed.* In the 16 papers presented and discussed by about 100 active participants in the three discussion sessions, the following subjects were prominent:

1. The use of nonnuclear primary elements for the measurement of conventional variables such as pressure, temperature, level, and flow in nuclear processes.
2. Measurements in high-intensity nuclear radiation environments of fluid pressures up to 3000 psig at temperatures to 600°F and fluid temperatures above 1000°F at pressures up to 200 psig (these ranges have since been extended).
3. Problems of measurement unique to slurry applications.
4. Instrumentation, control, and safety of in-pile experiments for the mutual operation and safety of the reactor and experiment.
5. The role and responsibility of the instrumentation and controls analyst.
6. Applications of apparatus such as data-handling equipment, alarm scanners, and data-logging devices.
7. Applications of transistors and magnetic amplifiers in process instrumentation.

A few conclusions were well developed at the meeting. The need for a continuing program of development and tests of primary sensing elements capable of operating at high temperatures and pressures for extended periods of time with little maintenance was apparent. New techniques, such as the use of eddy currents and ultrasound, need further evaluation. Instruments of high accuracy will have to be made available to aid in component development as the industry comes closer to economically competitive electric power generation. Similarly, long-lived instruments providing absolute containment of the process fluid, resistance to radiation damage, and provisions for remote adjustment, maintenance, and removal must be made available for nuclear plant service.

Because reactor systems are rapidly increasing in complexity, it was concluded that increased effort must be made to coordinate more closely the efforts of the designers and the controls specialists. The nuclear-power-plant designer has tended to do most of his thinking in terms of steady-state behavior of the system. He usually knows what the system is supposed to do in moving from one steady state to another, provided that the steady-state conditions required have themselves been well defined. Sometimes, however, analysis shows that the system design is such that no amount of instrumentation and control could prevent system damage. For example, a transient temperature excursion may damage a system designed so that relatively simple devices are adequate for steady-state operation. Similarly, the controls specialist may find that transport lags, poor sensor locations, etc., serve to give such unreliable information that frequent system shutdowns and attendant thermal cycling may take place. He must then work with the designer in an effort to modify the system. It is the reviewer's opinion that these situations are noted later than they should be in many cases. When the designs have already been transformed into hardware, desirable system changes may become quite costly, and compromises resulting in mediocre performance are too often the result.

* Although the conference was held in May 1958, the proceedings were not available until April 1960.

Greater usage of the techniques of redundancy, coincidence, and built-in testing was predicted for the next few years. Proper application of such techniques will increase the safety and serviceability of a nuclear reactor. Heretofore, efforts to design reliable control systems have frequently resulted in the use of a large number of protective devices designed to be "fail-safe" in as many respects as possible. The result (perhaps inevitable) has been an excessive number of cases in which the reactor was shut down because of minor troubles in the electrical circuits and instruments. The principle of redundancy will provide additional safety, as well as continuity of operation, in reactor protective systems. The use of two-out-of-three or two-out-of-four coincidence connection of instruments allows the use of built-in testing. With coincidence instrumentation, at least two instruments must indicate an unsafe condition before the reactor is scrammed; such a condition can be simulated for each instrument in turn to determine that it and its corresponding control circuitry are operating properly. If testing capability is built into each control system at the sensor end, the reactor operator can, at any time, even during operation, ascertain that the entire channel is operational. These techniques must, of course, be applied carefully to be effective. The testing of channels must be done often enough to give confidence that a channel which has "failed-to-danger" may be detected and that the channel may be placed on manual scram. If this were not done, the system could then become a two-out-of-two system, as mentioned by Siddall,[2] who has published a particularly thorough evaluation of problems of instrument reliability.

Recognition was evident during the conference that both the safety and economic future of nuclear power will depend heavily on the parallel development of instruments and controls of good quality and their proper incorporation and use in reactor systems. (R. G. Affel)

Safety Instrumentation and the Human Element

The safety of most experimental nuclear reactors is inherently dependent upon instruments and instrument systems, and the operating reliability of these instruments is, in turn, dependent upon their design, fabrication, operation, and maintenance, each of which is subject to human error. Since the instruments and associated systems may be handled by a single individual at each stage of this process, the same error may be applied equally to both single and multiple instrument systems. Human errors may fall into any or all of the following categories: (1) inadvertent action, (2) action with lack of knowledge, and (3) deliberate action. These errors can be minimized by improving the knowledge and thus the attitudes of all personnel associated with control and safety systems and by tightening administrative control of all persons associated with operation of the nuclear system. The intent of this article is to simply tabulate some of the human errors which have been encountered in the reviewer's particular experience and which substantiate the need for recognizing and improving "human engineering" in safety instrumentation in order to reduce the frequency of errors.

Many of the safety instruments in use at present are of the null-balance potentiometer type which incorporate safety action by means of cam-operated switches actuated by the motion of the balance system. The safety systems incorporating them can thus be negated by stopping the motion of the switch-actuating cams. This is most readily accomplished by manually deenergizing the instrument. Even though the loss of power to the instrument results in audible annunciation, this does not prevent the blocking of the safety-circuit action. For example, in one instance, while the operator was attempting to deenergize the suspected inoperative channel of a dual-channel safety system, an adjacent operating channel was inadvertently turned off and blocked. In another case the instrument was inhibited by physically blocking the balance mechanism so that it could not reach the safety limits. In certain instances instruments have been desensitized by incorrect adjustment (or improper replacement) of the standardizing mechanism, standard cells, control switches, balancing bridge, or amplifier components. In another instance an untrained individual opened the bridge circuit to an instrument while attempting to change the recorder chart. Certain of these errors may remain undetected until a thorough instrument check is conducted.

The extent to which operator error can impose danger to reactors and personnel is particularly pertinent to interconnections between

experiments and reactors.[3] In a research reactor, control signals from an in-pile experiment enter the reactor controls system through a key-locked isolation switch. This switch was installed to assist in the experiment checkout without activating the reactor circuits. The intent of this switch was abused a number of times when it was actuated during reactor operation to permit correction of a malfunction in the experiment instrumentation. Thus, in an effort to correct a minor problem, all experiment control was disconnected from the reactor, a purpose for which the switch was obviously not intended to be used.

Primary sensing elements with multiple signal and power lead connections have been deactivated by the inadvertent crossing of the leads. It is particularly difficult to detect a malfunctioning sensing element, since, in most instances, it is impractical to generate an appropriate test signal at the sensor.

The final control element is also subject to human error, as illustrated by the interchanging of control-valve signal lines and the interchanging of three-phase rotating machine power lines. These errors reverse the aspect of control actions and thereby force the control or safety system to take positive action in the wrong direction. Such errors are usually detected, however, by routine checking procedures.

Similarly, system coolants, both gaseous and liquid, have been jeopardized by misconnection of the supply source. In some cases the fault has remained undetected until other system actions have gone beyond safety limits.

Frequently an error is made during the construction of an auxiliary system after operation of the primary system has begun. An error may readily occur in any of the transmission media at virtually any place in the safety system in such a situation, including previously checked components.

The physical blocking of spring-return push buttons, relay contacts, etc., has been accomplished with matches, pencils, and similar foreign materials. Switch action has been inhibited by deposits of oil, wax, and other chemical films on contacts.

An insidious error has been detected in regard to the position of a flux monitoring chamber with respect to the reactor. Since the chambers are only relative, rather than absolute, measuring devices, their positions are very critical and should be so honored. However, changing reactor conditions (fuel burnup, poisons, etc.) do require chamber repositioning. It should suffice to say that all chambers must not be moved at any one time, and proper care must be exercised to ensure that the designated chamber is, in fact, the only one repositioned.

The hidden circuit jumpers that have been installed for testing purposes prior to operation and then left intact are the most difficult errors to detect. The reviewer considers these to be the most serious errors encountered. Frequently such errors can only be detected via a wire by wire check, for circuit malfunction may occur only after a number of sequential conditions are established. Temporary electronic circuit modifications that have been made for different modes of operation pose the same problem in that they are not easily detected, particularly if they are of the level-limiting type.

Of particular interest are the number of failures caused by the inadvertent plugging of coolant and ventilating lines. Most of the items (rubber stoppers, cotton padding, clothing, and weld slag) enter the system during fabrication and may not be detected during component checkout. Fortunately these errors are usually found as a result of malfunctions during the system checkout, and serious situations that could have occurred during power operation are avoided. Frequently the only monitor for such a flow system is of the absolute pressure type which will not necessarily indicate a loss of flow, as would a differential pressure sensor.

Human errors in the safety instrumentation, such as those previously described, will be discovered during component checking prior to operation of the system, during system checking prior to operation with nuclear power, or when a malfunction of the system occurs during operation. To date, those errors which have escaped detection during either the component or system checking prior to operation have become apparent through action from a correctly operating safety channel. The safety system of a single process should incorporate a minimum of two independent channels utilizing separate control variables, each having equal and satisfactory degrees of reliability. The number of human errors can be materially reduced by the means described above, but the scope of a single error can be limited only by some form of

redundant control. A system that utilizes only one control parameter for control and/or safety should be viewed with extreme caution, especially if the detection and control channels are identical, even though redundant.

Finally, in engineering to counteract the foibles of the human element in safety instrumentation, two factors are of considerable importance. First, regardless of the type or number of safety systems employed, a paramount requirement is that the entire complex be built upon a foundation of accurate documentation. Since experimental reactor-control schemes change frequently, it is imperative that formalized documentation be continually updated so as to preserve an accurate, intelligent account. This inherently requires clear rules and principles for operating, testing, and maintaining instruments and controls involving safety. Second, regardless of the types of human element control which may be selected, it is imperative that clear lines of responsibility for all phases of operation be established and actively perpetuated. (K. W. West)

Fuel-Element Leak Detection

The basic methods for detecting and locating burst fuel elements in nuclear reactors during operation were discussed in the September 1960 issue of *Nuclear Safety*.[4] No new concepts for devices have been reported since that time, but some recent developmental work is discussed here.

A simple monitor for gas-cooled reactors and in-pile loops has been suggested by Heath[5] which consists of two filters that are loaded with CWS No. 6 paper and are placed in series within a cooled gas-sampling line. Particulate matter, solid fission products, and their solid daughters are collected by the first trap. Only the coolant and the noble-gas fission products pass through this type of filter paper, and even a high percentage of molecular-size particles are chemically adsorbed. The second filter is placed an appropriate delay time downstream and is periodically monitored by a scintillation spectrometer for adsorbed rubidium and cesium, the decay daughters of krypton and xenon that appear in the time between passage of the coolant through the first filter and the second filter. A comparison of the count rates obtained at suitable time intervals yields an indication of the presence of a leaking element.

One restriction on the use of this device is that the sample effluent must be cooled to well below the ignition point of the filter paper. A continuous paper-tape monitor can be adapted for use with this trapping system. Since the tape would not be appreciably contaminated until a leak occurred, the tape could be spliced into a belt. The extreme simplicity makes this system appear to be very desirable for programed monitoring of multichannel gas-cooled reactors.

Preliminary developmental work has been done at Battelle Memorial Institute (BMI) on a silver bromide–loaded ion-exchange column that can be monitored for delayed-neutron emission from adsorbed precursors of iodine and bromine.[6] This system appears to be a neutron analogue of the Heath ion-exchange gamma-ray monitor widely used in water-cooled reactors.[7] The delayed neutrons from O^{17} and photoneutrons would contribute to the background of this monitor in a high-flux reactor.

Direct delayed-neutron monitoring is being considered for the organic-moderated reactor at Piqua[8] and has been tested for use with water-moderated reactors at Chalk River.[9] The reaction

$$O^{17}(n,p)N^{17} \xrightarrow[4.14\ \text{sec}]{\beta} O^{17} \rightarrow n + O^{16}$$

from the capture by natural O^{17} in water and other oxygen-containing coolants must be taken into account in this type of monitoring in that this contribution to the background must be attenuated by delay lines. The use of delayed-neutron monitoring of NaK- and sodium-cooled reactors is also being studied.[10]

Design data for gamma monitors[11] and for systems for locating leaks[12,13] have also appeared in the current literature. Actual operating experience with fuel-element leak-detection devices has not, however, been reported, except in rare cases, and the experience has been too inconclusive to permit comparisons of the various devices. (J. L. Kaufman)

Reactor Fuses

Reactor fuses are safety devices that automatically insert poison into the reactor when

triggered by some phenomenon associated with a reactor excursion. An article in the September 1959 issue of *Nuclear Safety*[14] described the devices being developed in the BORAX, SPERT, and aircraft-nuclear-propulsion programs, and the current literature indicates continued activity in the development of reactor fuses. Developments in this area are of particular interest to those charged with responsibility for reactor safety. These individuals, at one time or another, will inevitably synthesize the ideal safety device, and it will probably be a fuse. Effective reduction of present concepts to practice is the major current problem. The following discussion reviews work in this area.

A Coupled-Diaphragm Safety Device for Power Reactors

The development of a reactor fuse intended for use in power reactors is described in two Atomics International reports.[15,16] The basic principle employed in the trigger is shown in Fig. III-1. Balanced pressure is used so that

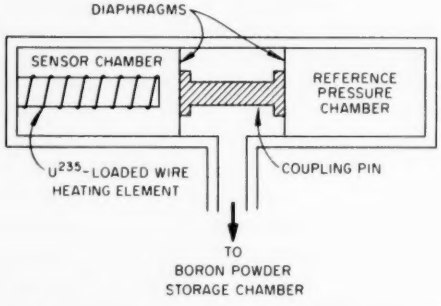

Fig. III-1 A coupled-diaphragm safety device for a power reactor.

environmental temperature changes will not alter the load on the diaphragms, and thus considerable latitude is allowed in applying the fuse. The fuse operates when excess pressure produced in the sensor chamber by heating of the uranium-loaded heating element ruptures a diaphragm and releases high-pressure gas into the boron powder storage chamber. As the gas passes into the main body of the fuse, it picks up boron powder and disperses it into a region of higher flux.

It is evident that considerable effort has been expended in developing the final design used in the prototype. Numerous diaphragm-burst tests were performed in which consistent results were obtained, and then a prototype was tested in which the sensor chamber heating was simulated with a resistance winding connected to an exponential generator. The results of the test are presented in Table III-1 in terms of time response of the fuse.

Table III-1 TIME RESPONSE OF A PROTOTYPE COUPLED-DIAPHRAGM REACTOR FUSE

Event	Elapsed time, sec
Exponential power increase started, 0.200-sec period	0
Trigger tripped	1.10
Powder released	1.15
Powder moved 42.5 in. down in receiver	1.19
Powder fully dispersed	1.23*

* Estimated time for dispersion of powder by a device of the size required for the Sodium Reactor Experiment (SRE).

As pointed out,[16] the fuse is intended primarily for power reactors. In any reactor the time response of the safety system must be that necessary and sufficient to deal with the maximum credible reactivity insertion rate. This is often relatively slow if compared with the time response required to provide safety action in small reactors with short neutron lifetimes.

The reviewer suggests that fuses intended for use in large, servo-controlled power reactors be equipped with means for indicating whether the fuse has been actuated. In such applications it is likely that a large number of fuses will be used and that the negative reactivity contributed by the misoperation of any particular fuse will be compensated for by the servo and thus not called to the attention of the operator.

Fast-Period Reactor Fuses

Two fast-period fuses, a large annular-passage fuse and a control-rod type fuse, are described in a recent APEX report,[17] and the annular-passage fuse is described in more detail in reference 18. The control-rod fuse is not wholly self-contained in that an external power source is used to initiate poison insertion. These devices are safety rods characterized by the extremely rapid and irreversible insertion of nuclear poison in the form of gas

or finely dispersed powder. Although considerable information is provided on the performances of the fuses, it is not possible to fully assess the data in terms of the protection thus provided. The operation of these fuses is extremely sensitive to the local environment at the thermostat, and their effectiveness is sensitive to location in the reactor; therefore their application in any reactor system will require more than the usual amount of care.

Reduced to essentials, the devices consist of an evacuated tube containing, at one end, a concentration of nuclear poison, an explosive charge, and an electric detonator triggered by a nuclear thermostat. The design is such that, ideally, the poison assumes hollow-cylinder geometry after insertion. The sensing element is a bimetallic snap-action-disk thermostat composed of uranium and zirconium. These metals have extremely low thermal and mechanical inertia and excellent response. It is stated that "Detonation, caused by completion of the circuit, is accomplished less than 1 millisecond after the disc gets ready to snap." In the tests the thermostat was located in the high-flux region of the safety rod to obtain maximum sensitivity.

The peak power attained in the SPERT-I reactor with four small control-rod fuses located at the corners of the lattice was two decades less than the self-shutdown peak power with the reactor on a 0.024-sec period, although, as would be expected, the development model thermostats varied considerably in their operating temperatures. During a transient, the fuse temperature is proportional to the energy developed in the reactor; i.e., the trip-point power will be inversely proportional to the reactor period. The detailed analysis of poison insertion versus time shows that less than one-half of the total reactivity of the fuses was inserted at the time peak excursion power was attained. This emphasizes that the first duty of any safety system is the immediate removal of a relatively small amount, the prompt excess, of reactivity.

The report[17] states that the thermostat temperature can rise from 10 to 50 times as fast as the temperature in some fuel elements in the same flux. The reviewer presumes that these figures are derived from Eqs. 2 and 3. It must be pointed out that these are average temperatures and that they become less meaningful with decreasing reactor periods. Thermostat response might show improvement, however, at shorter periods because of larger differential temperatures between the uranium and the zirconium. The report implies that reactor damage produced in an excursion is a simple function of the energy developed. The reviewer suggests that this is not necessarily the case. Thermal stresses produced by large temperature gradients in fuel elements will be a significant factor in the damage mechanism, and these are strongly dependent on period. Safety system performance should not, therefore, be designed with developed energy as the sole criterion.

The report[17] includes a well-considered examination of the problems of thermostat location. These are similar to the problems of neutron chamber location in more conventional systems. Since the thermostats may or may not be located in the safety rod proper and since electrical interconnections are required between the primer, the thermostat, and the detonator power supply, a monitoring system designed to annunciate a loss of system continuity is required. The reviewer suggests, further, that the existence of the detonator voltage across the thermostat be monitored so that the existence of high contact resistance across the contact of the thermostat, which could prevent sufficient current flow, could be detected. The report points out that the use of multiple fuses and actuating thermostats in parallel will effect an increase in the probability that circuit closure will be attained, but only one power supply, a battery, is provided to serve the multiple array.

The report[17] concludes with an informative summary that briefly discusses some of the many approaches taken in designing fuses and fuse triggers. The opinion is expressed that a radiation-sensitive trigger, namely, an ion chamber or a corona-sensitive discharge tube, would prove superior to a thermostat. The inclusion of such a trigger would produce a system whose static configuration would be virtually identical to that of the conventional shim safety rod, the principal differences being that a powder charge is used to provide poison insertion and that the device is not usable as a shim. Such a design might lose the advantage of the small size and simplicity of the thermostat.

The report[17] points out that these fuses, as designed, are intended for use only at zero or low power levels during initial experiments and that the over-all response time of these fuses is reported to be ~0.01 sec. Radiation damage

is not a factor in the design. If the thermostat-operated fuse was designed to be used in the power region and located to attain the desired fast response, radiation damage to electrical insulation would undoubtedly be a serious problem. It is assumed that, in any application, the tissue paper used to retain the poison powder would be replaced by a more suitable material.

(J. R. Tallackson)

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Section IV

PLANT SAFETY FEATURES

Underground Containment of Power Reactors

A nuclear power reactor operating for an extended period of time will build up a sizable fission-product inventory that could cause disastrous loss of life and property if released. For this reason, most power reactors are provided with a containment envelope independent of the rest of the power system so that failure of the system will not result in excessive activity release. The normal procedure[1] is to provide a vapor-containment pressure vessel, designed to low-leakage specifications, along with an exclusion area surrounding the plant, which prevents inhabitants of areas near the plant from being exposed to excessive radiation in case of an accident. The exclusion area is needed to reduce the direct gamma irradiation at the boundary site and to allow sufficient air dilution of fission products leaking from the containment vessel.

In some countries the purchase of exclusion areas is extremely expensive, and in some cases it is difficult to find acceptable sites that are sufficiently distant from large inhabited areas. This is partly because thermal power plants require a heat sink, which usually stipulates a site near a river, and the bulk of the population is usually settled along rivers. In Switzerland, for instance, if a 10-km circle is drawn around each settlement with a population greater than 10,000 inhabitants, it is found that almost all the river sites are overlapped.[2] Thus any scheme that eliminates the necessity of an exclusion area would be advantageous in this case. Furthermore, land acquisition prices would be reduced, along with power transmission costs. Underground construction of nuclear power plants could provide a solution to this problem, even though the actual cost of such construction would usually be higher.

Although in more sparsely settled countries it would be more expensive to build a plant underground, almost complete invulnerability to military attack could be attained. This facet of the problem is usually overlooked when decisions are made on a strictly economic basis, but the fact remains that the supply of power is one of the prime prerequisites to reconstruction after military destruction.

Underground Construction

Underground construction, as discussed here, specifically refers to installation of the power plant in a cave or in excavated rock. There has been a great deal of experience on tunneling and other facets of underground construction. Norway has actually built the Halden nuclear plant in a rock excavation,[3] and many hydroelectric power plants have been built underground on a strictly economic basis. From a technical point of view, there are several advantages of a rock-contained plant. Among the advantages is the fact that the internal pressure in the chamber caused by an accident would be taken up wholly or, at least, in part by the rock. The reactor chamber could be readily located so that no fragments of an exploded plant could reach the surface of the ground and thus create a direct path for the release of fission products to the atmosphere. The possibility of destruction of the plant from without would be decreased. Fission products in the form of gases or aerosols would have to seep through rock before being able to reach the surface of the ground, and in the process they would lose activity since (1) there would be natural decay during the time required to seep through the

rock, (2) some fission products would be retained in the rock and in the overburden soil by ion exchange, (3) aerosols and vapors would be arrested on damp surfaces, and (4) inert gases, such as xenon or krypton, would decay after a relatively short period of time into products that could be absorbed by the rock and the soil overburden.

However, there are disadvantages. The costs of underground installations, although they may be competitive in foreign countries, have been estimated to be from 3 to 7 per cent of the total plant cost greater than the costs of an equivalent surface installation.[4] Further, it is not always possible to ensure ahead of time that the rock quality will be good throughout the volume of excavation and that heavy flow of ground water does not exist; also it is not always possible to find an acceptable geological structure adjacent to a large body of water. Cooling towers could be used, but these are expensive and would have to be placed outside the protective area. Ground water may be contaminated in case of an accident concurrent with rock shifting or a slight movement which might cause a fissure of the excavation lining. Finally, there would be limited flexibility with respect to access and expansion. A nontechnical advantage of the underground installation is that the surface topography would be unchanged and would thus maintain its natural beauty.

Some unique requirements must be met for a site to be acceptable for underground construction. The rock formation should be easy to excavate but should be structurally sound. It would be advantageous if the topography were such that a horizontal entry could be made, i.e., from a side of a hill. Existing openings, such as mines and caves, could be used at a great saving in cost if they were adaptable to the requirements. Such sites have been surveyed for underground installations.[4]

It appears that limestone, granite, sandstone, and salt formations have the best properties for excavation.[4] These formations usually permit wide spans, high ceilings, and regular pillar arrangement. Limestone is the primary geological formation used in the United States for underground installations. The limestone deposits in the United States are generally horizontally bedded, and many sites are desirable because they permit horizontal entry, large flat roof spans, and high ceilings. Granite gneiss deposits are second in importance. Most of the investigated granite formations are accessible by horizontal entry from the sides of hills or cliffs. Granite formations usually permit large arched roof spans and high ceilings. Sandstone formations are in many cases adaptable for underground construction, but possible sites are not so abundant as those in limestone or granite. The average strength is less for sandstone than for limestone or granite, and its porosity is much higher. Salt deposits are favorable for underground construction and have the added advantage of dryness. However, entry must be by means of deep shafts. Where cost is not important, a plant could be installed in virtually any type of geological formation.

Excavations are usually made by drilling and blasting. The procedure is to drill a vertical entry to the top of the planned chamber. The roof section is excavated first, and "dental work" repairs are made to the roof to assure structural integrity. This work consists of tying poor rock together by steel straps, bolts, and concrete and, if called for, lining the excavation with concrete. Once this has been accomplished, excavation can proceed downward to the required depth without the hazard of rockfalls from the ceiling.

There is a great deal of experience in underground construction of this type. As of 1957, over 240 hydroelectric powerhouses had been built underground. Of these, the greatest number are in mountainous countries with reasonably advanced technology and relatively low labor rates; for instance, 44 are in Sweden, 53 in Norway, 36 in Italy, and 28 in Switzerland.[5] The United States, however, has only four plants of this type but has had much experience in tunnels, mining, and other forms of excavation.

Containment in Underground Plants

The containment structure enveloping a nuclear reactor must withstand various loads that are not dependent on whether the plant is aboveground or underground. For example, the internal pressure that would result from a loss-of-coolant accident or accidental chemical reactions in a pressurized-water or gas-cooled reactor plant would have to be successfully resisted by the structure. The escape of fission products would have to be prevented, as well as direct radiation damage to the nearby population, and measures would have to be taken to prevent breach of the enclosure by missiles.

The underground plant can be constructed to satisfy each of these requirements. Leakage of radioactivity could be positively prevented by a steel-sheet lining, or it could be reduced by various schemes. The chamber could be lined with concrete sprayed directly against the rock after excavation, in which case permeability of the concrete and overlying rock and the ion-exchange properties of the rock and overburden would be the determining factors in the release of gaseous contamination. Ground-water contamination could be controlled by placing drain tile around the reactor excavation leading to holdup tanks.[6] This would reduce the water table in the vicinity of the reactor and provide for control of the release of radioactivity. A steel or plastic liner could be embedded in the concrete lining the rock walls, but unless the inner surface of the concrete was further lined with a nonabsorbing or removable substance, decontamination in the event of a minor accident would be difficult. The decontamination of bare rock would, of course, also be difficult.

An interesting scheme[2] is to line the rock walls with a course of load-bearing hollow blocks and to line the blocks with concrete. The annular space thus provided could then be pressurized so that leakage would be maintained into the enclosure from which it could be sent through filters and then discharged. The problems inherent in this system are similar to those of conventional building containment.[7]

If reliance is placed on the holdup time of the fission gases in the rock and ion exchange in the rocks and the soil, good knowledge of the underground water flow is required to avoid contamination of human water supplies. Absorption of fission products in various clays and rocks has been reported,[2] and the general problem was discussed in an earlier issue of *Nuclear Safety*.[8] The relative effectiveness of the various materials in ion-exchange absorption of Sr^{90} is indicated in Fig. IV-1 (taken* from reference 2), in which the quantity of absorbed strontium in milligrams per 100 g of rock or clay is plotted against the initial concentration of strontium in the solution. These data were obtained by adding finely ground rock or clay to the solution and allowing it to react for 2½ hr. The weight of the rock in the test

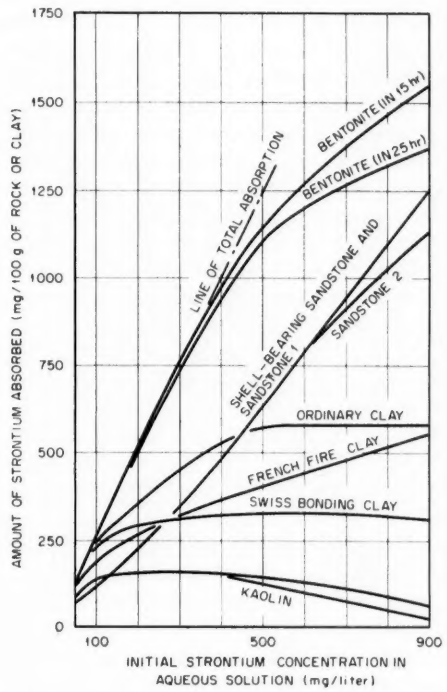

Fig. IV-1 Absorption of strontium in different clays and rocks.[2]

solutions was in the ratio 1 : 25. The strontium in solution was in the form of strontium nitrate. Figure IV-1 shows that the relative effectiveness in ion exchange of strontium is, for low concentrations, in decreasing order: bentonite, ordinary clay, Swiss bonding clay, French fire clay, kaolin, and sandstone. At higher concentrations, the order becomes: bentonite, sandstone, ordinary clay, French fire clay, Swiss bonding clay, and kaolin. With low strontium concentrations, complete absorption takes place with bentonite, whereas limestone absorbs less. Figure IV-2 (also taken from reference 2) shows similar data for the absorption of iodine in different clays and rocks. These data are for iodine in atomic form, dissolved in water with the addition of a small quantity of potassium iodide. Bentonite again shows the greatest absorption capacity. The data of both Figs. IV-1 and IV-2 indicate the hydrologic problems that exist in the field.

It is probable that a large absorption factor can be attained in natural rock if the crevices are not too large and, in addition, are moist.

*Figures IV-1 and IV-2 are reprinted[2] by permission from Comitato Nazionale per le Ricerche Nucleari, Rome.

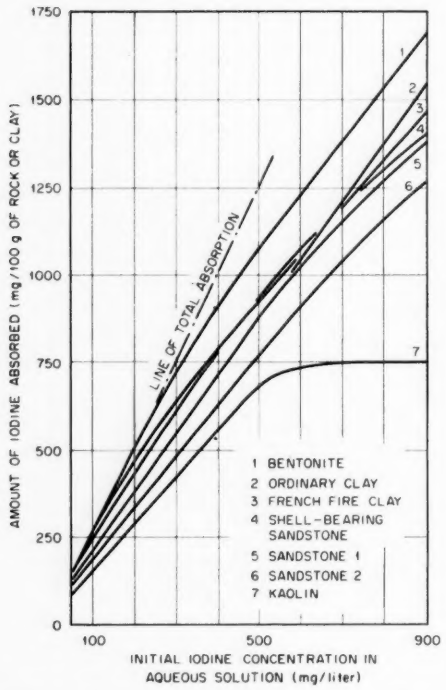

Fig. IV-2 Absorption of iodine in different clays and rocks.[2]

The structure can be kept moist by proper growth of vegetation on the overburden soil.[6]

Cost of Underground Containment

The cost of containment constitutes a considerable portion of the total cost of a reactor installation. A hypothetical underground design for containment of the EBWR has been reported by Beck,[4] in which the layout is in the same form as for the aboveground installation. In sound rock the 80-ft-diameter dome-shaped scheme should be feasible. For large openings in excess of 50,000 cu yd, Beck estimates the excavation cost to be possibly as low as $8 to $10 per cubic yard. For the reference EBWR design, it was estimated that the cost of the underground container would be approximately $940,000; this cost includes rock excavation, concrete linings, tunnels, ventilation, and blast-proofing features. The over-all cost of the underground plant was found to be 7 per cent greater than the cost of the aboveground plant, as built.

The actual cost of the Halden underground reactor building, including the pretunnel, in Norwegian kronen, was as follows:[3]

Excavation of a volume of 8900 m^3, including securing of rock surfaces before concreting	650,000
Concreting, including concrete shielding, air-lock walls, and painting	850,000
Total	1,500,000

The total civil-engineering cost amounted to 2,760,000 kronen. Aamodt[3] estimates that an equivalent steel-shell structure in Norway would cost 2,950,000 kronen. At the official exchange rate of $0.14 per kronen, this is about $386,400 for the underground construction of an 11,650-cu yd volume. The construction costs come to about $332 per cubic yard for the total work, including $7.85 per cubic yard for the excavation. Of course, it is difficult to relate this to actual costs in the United States because of different labor rates and amounts of mechanization. Aamodt[3] states that this is a relatively high cost because of the difficult transport of the excavated material from the lower parts of the plant, together with the rather extreme bolting of the cracked zones performed before the concreting. Beck,[4] however, considers a cost of $8 to $10 per cubic yard quite a low figure, attainable only with large excavations; the actual cost in the United States at the present time probably varies between $10 and $30 per cubic yard.

Conclusions

It is clear that underground construction of reactors is feasible and, in some areas, may be competitive with aboveground construction of a steel containment shell. As an exclusion area becomes increasingly valuable, underground construction becomes more attractive. At the present time, it could be justified in some cases on the basis of invulnerability to military attack. Since, as mentioned above, a supply of power is one of the first requirements for reconstruction after a disaster, the 3 to 7 per cent increase in cost over that of the aboveground plant could be considered as insurance against total disaster. (M. H. Fontana)

Removal of Radioiodine from Reactor Gases

Provisions for control of the radioiodine produced in a nuclear reactor are of special importance because this biologically hazardous material is volatile and thus may be released from nuclear fuel under certain conditions. The biological effects of radioiodine are such that a very low maximum permissible concentration in air has been established.

As a result of the importance of the control of radioiodine, its removal from gases has been the subject of a number of recent articles. A review of the pertinent information published prior to October 1959 was presented in an earlier issue of *Nuclear Safety*.[9] The earlier review covered the removal of radioiodine from gas streams by caustic scrubbers, by heated silver nitrate reductors, by silver-plated metal mesh, by activated charcoal, and by other solid adsorbers.

Removal-System Design

Geldart[10] has since reviewed the literature and has illustrated the principles of design of an iodine-removal system. Further, by application to a hypothetical problem, he has evaluated the various iodine-removal methods. The effect of condensation nuclei in the gas stream on the behavior of iodine-removal material is noted. Of considerable importance is his selection of a system that is always ready for operation and does not require the activation of pumps or heaters. The example selected to illustrate the design of an iodine-removal system was an in-pile loop using high-pressure CO_2 coolant at 600°C. In the event of accidental release of radioactivity to the coolant, the gas would be discharged through a heat sink consisting of a bed of graphite chips or steel ball bearings, where it would be cooled to less than 100°C. The gas would then pass through a prefilter and an absolute filter, followed by a bed of activated carbon, and, finally, through another filter that would capture any carbon dust with iodine attached.

Sorption on Materials in Gas Streams

The sorption of iodine on solids that could be carried in the effluent gas from a graphite-moderated air-cooled reactor and desorption from those solids have been investigated by Hudswall et al.[11] Their tests were designed to determine the fate of iodine released from burning uranium metal rather than to measure the efficiency of prospective filtering materials. The reactor designer can use information of this nature for predicting the distribution of radioiodine in a gas system. In these experiments the radioiodine was introduced into an air stream flowing at the rate of 500 cm^3/min by burning irradiated uranium wire. The air was passed through a sintered glass disk and then through a thin bed of about 5 g of the solids being tested. The bed of solid material was held at 200°C. After measurement of the amount of iodine retained by the test material, about 0.5 mg of normal iodine was introduced into the air stream in order to ascertain whether there would be displacement of the radioactive iodine from the test solids by exchange.

The data show that 90 per cent or more of the iodine was removed from the gas stream by the following materials: magnesium, lead, bismuth, copper, graphite, MgO, PbO, Pb_3O_4, U_3O_8, Al_2O_3, and concrete dust. Other materials tested included Bi_2O_3, which removed 80 to 95 per cent of the iodine; UO_2, which removed 85 per cent; and filters, identified as "Windscale type," which removed 40 to 45 per cent.

Negligible amounts of iodine were released by exchange from bismuth, MgO, and the dry (i.e., unlubricated) Windscale type filters. The amounts of iodine released by exchange from the other materials ranged from 10 to 75 per cent. Copper, PbO, and concrete dust were not tested in these exchange experiments.

In the desorption tests, iodine release from PbO was initiated at 400°C; release from MgO started at 700°C; and the filter material gave off iodine at 250°C. Graphite did not release iodine while being held at temperatures up to 350°C for ¾ hr, but it began to release iodine at 450°C. The retention of iodine by graphite was not affected by the presence of air at temperatures below 450°C. Magnesium oxide continued to retain iodine after 1 hr at 600°C in addition to 1 hr at 500°C and 1 hr at 400°C. In other tests the adsorption of iodine by these solids from carbon tetrachloride–iodine solutions was measured. Magnesium, MgO, graphite, and concrete dust exhibited the greatest affinity for iodine under these conditions.

It was concluded from these tests[11] that adsorption of iodine by solids that could be carried

in the effluent gas could explain the transport of iodine. Further, the collection of these solids by stack filters could result in partial retention of iodine after an accidental release of activity.

Removal from Steam-Air Mixtures

Many nuclear reactors have large quantities of water within their containment enclosures. In such cases an accident involving the release of radioactivity might also release large quantities of steam or water spray into the atmosphere as the result of heating of the coolant or pool water or the discharge of hot, pressurized water. In order to design a system for removing iodine from the gas in a containment enclosure under such conditions, information is required on the efficiency of the various iodine-removal systems in hot, steamy air. Two reports on this subject have recently been issued.

Adams and Browning[12] have tested the efficiency of activated charcoal for removing iodine from steam-air mixtures in laboratory experiments at temperatures ranging from 75 to 120°C. The air saturated with steam was used in tests at temperatures up to 100°C, and, in tests above that temperature, pure steam at a pressure of 1 atm was used, along with a small amount of air to maintain oxidizing conditions. The experimental arrangement was similar to that used in earlier tests.[13]

In the recent tests, the gas was passed through a $\frac{3}{4}$-in.-thick charcoal adsorber for a 2-hr period, which is the estimated time required to remove most of the suspended iodine from the gas in a containment building. Removal efficiencies of 99.9 per cent were observed at gas velocities of 25 to 75 ft/min; in a single test at a gas velocity of 290 ft/min, however, the efficiency dropped to 99.5 per cent. In similar tests at 100°C with air alone, at a velocity of 50 ft/min, removal efficiencies of 99.95 per cent were observed.

Apparently water vapor at 100°C has little effect on removal efficiencies. However, the effect of temperature on the removal efficiency is significant. The removal efficiency of 99.9 per cent obtained at 100°C is a factor of 10 lower than that obtained in the earlier tests at room temperature.[13]

Pilot-plant-scale tests of several methods of removing trace quantities of radioiodine from air streams have been conducted by Arthur D. Little, Inc. (ADL), for the Hanford Atomic Products Operation of the General Electric Company.[14] Iodine-removal efficiencies were determined for a silver-plated copper-mesh bed 2 in. deep and for beds of activated coconut charcoal, Linde 13X molecular sieve, and sodium thiosulfate, each $\frac{1}{2}$ in. deep, at temperatures of 70 and 160°F and at relative humidities at each of these temperatures of 70 and 95 per cent. Iodine concentrations of 0.01, 0.10, and 1.0 ppm were used in air flowing at a rate of 4000 cfm. The iodine was labeled with I^{131} and was injected into the air stream by continuously burning a metered stream of alcohol containing the labeled iodine. Evidence that most of the iodine produced by the flame was in the form of elemental iodine was obtained by analyzing the iodine content of the gas during a special run with a standard thiosulfate solution. Agreement with the amount injected to within 2.7 per cent indicated that not more than 2.7 per cent of the iodine could be in other than the elemental form. The efficiency of iodine collection was determined by withdrawing 0.1 per cent gas samples at points before and after the collector being evaluated. The iodine was removed from the gas samples by sodium thiosulfate–sodium hydroxide scrubbing columns. The solutions in these columns were monitored with a scintillation counter. The air was passed through a prefilter and an absolute filter after injection of iodine and before entering the iodine collector being tested. Since these filters remove iodine from the air stream, the concentration of iodine in the air entering the iodine collector was 5 to 20 times lower than the nominal concentration. The ranges and averages of iodine-removal efficiency observed are given below:

Collector	Iodine-removal efficiency	
	Range, %	Average, %
Silver-plated copper mesh	8–99.99	61
Activated charcoal	67–99.99	94
Molecular sieve	30–99.99	62
Sodium thiosulfate	24–84	51

In special tests a substantial decrease in the iodine-removal efficiency of silver-plated copper mesh was noted when 12 g of iodine had been accumulated in a bed 2 ft square and 2 in. thick. In a similar test with the activated-charcoal bed, no significant change in iodine-removal

efficiency occurred even after 1035 g of iodine had been accumulated in a bed that was 2 ft square, ½ in. thick, and pleated in a frame 8¾ in. deep. There was no significant difference in the efficiency of the activated-charcoal bed in the tests at 70 and 160°F at 95 per cent relative humidity. The molecular-sieve iodine collectors exhibited a high efficiency for a brief period, but their effectiveness deteriorated rapidly as they became loaded with water. The sodium thiosulfate collectors did not attain a high efficiency and failed after a short testing period. Apparently the thiosulfate was leached out of the bed by condensed water. Difficulty was encountered with one of the silver-plated copper-mesh collectors when an attempt was made to remove an oil film from its surface. Tar used in the manufacturing process then migrated to the silver surface and reduced its efficiency.

The efficiency of particulate filters for removing iodine from the gas stream was determined incidental to the tests described above. The efficiency of prefilters was quite erratic, varying from 0 to over 90 per cent. The efficiency of one of the Fiberglas absolute filters fluctuated between 70 and 90 per cent and gradually decreased to less than 10 per cent as the accumulation of iodine on the filter increased to 200 g. Another absolute filter of the same type fluctuated widely in efficiency between 0 and over 90 per cent. The conclusion drawn from these tests was that the activated-charcoal iodine collectors were the only ones that continually showed acceptable iodine-removal efficiency.

It is of interest to compare the results reported for charcoal by ADL and by ORNL. Although the performance of activated charcoal in the ADL test was satisfactory for the intended use, there was a significant difference between the results of the ADL and the ORNL tests. The ADL tests gave average efficiencies of approximately 94 per cent, whereas the ORNL tests gave efficiencies of 99.9 per cent or more. The experiments differed in that different bed temperatures were used; also the ADL adsorber bed was ½ in. thick, whereas the ORNL bed was ¾ in. thick. These differences are not, however, sufficient to account for the different results.

The ADL tests were run in a large pilot plant, and the charcoal beds were delivered to ADL as a black box. The internal construction of the box could not be inspected. The manufacturer's instructions called for tumbling the assembly to eliminate channels that might have appeared in the charcoal bed during shipping. Since the interior of the collector assembly could not be inspected, it was possible that vacant channels could have been left in the adsorber.

One explanation for the wide variation of iodine-collection efficiency of the particulate filters is that there may have been variations in the particulate content of the air at that point in the system. Prefilters were used to reduce the dust content of the entering air to below 0.005 mg/meter. Since the entering air was drawn from outdoors in a metropolitan industrial area, it is conceivable that condensation nuclei in varying amounts could have been present which could have penetrated the prefilters and produced the varying results observed with the absolute filters as well as with the iodine collectors. It is apparent that, if the iodine-removal efficiencies of 99.9 per cent or more observed in laboratory experiments are to be attained in full-scale installations, it will be necessary to provide for inspection and control of operating conditions to ensure that conditions similar to those tested in a laboratory are maintained.

Test of an Installed System

In laboratory-scale or pilot-plant-scale tests of the removal of iodine from gases, an effort is made to duplicate all the significant conditions that will exist in the system to which the results of the test are to be applied. There is always a possibility that one or more factors may be overlooked, and therefore it is of interest to conduct tests on actual installations from which radioactive iodine might be released. Such an experiment has been reported by Chamberlain et al.[15] In this experiment, iodine containing I^{132} was released into the containment shell of the PLUTO reactor, and the behavior of the iodine in the containment shell and in the iodine scrubber was observed. Parallel beds of two different solid adsorbers were added for comparison. The three scrubbers tested were (1) a Peabody scrubber containing sodium carbonate solution, (2) a 6-ft-deep 3-in.-diameter copper-mesh bed made of wire 0.25 mm in diameter with a bulk density of 15 lb/cu ft, and (3) a 10-in.-deep 6-ft-diameter activated-charcoal bed containing 6- to 10-mesh charcoal. Paper

absolute filters were installed on the upstream side of the iodine collectors, and an air flow rate of 100 m^3/min was used. Three runs were made with a different iodine collector in the circuit each time. The initial iodine concentration in the containment shell was about 1 $\mu g/m^3$. The air was continuously recycled, and sampling was carried out periodically over 5-hr periods. The scrubber and the copper mesh contributed little to the over-all decontamination, whereas the charcoal yielded a decontamination factor of 50 to 100 in the 3 hr during which it was tested. Of the iodine released within the containment shell, 90 per cent was deposited within the shell, 10 per cent was eventually absorbed in the decontamination plant, and 0.5 per cent remained more or less permanently airborne and was not removed by the copper or scrubber systems. The charcoal system removed this fraction with a decontamination factor of at least 20. It was conjectured that some of the iodine in the system had reacted with trace impurities to form at least two different compounds, one of which is adsorbed by charcoal and not by the scrubber or copper, whereas the other compound is not absorbed by any of the systems. It was also conjectured that the latter compound might be such a large molecule that it would be equivalent to a small particle, which would be difficult to filter out and would also have poor diffusivity.

Effect of Form of Iodine

It is noted in several of the reports discussed in this review that the form in which the iodine appears affects its behavior in the collector systems. In some cases the different forms of iodine could be identified and associated with the variation in collection efficiency,[10,13,15] and in other cases the presence of such effects could be inferred. In the various systems in which iodine collectors are expected to operate, there may be a wide variety of conditions and impurities in the gas which could affect the form of the iodine. In addition to the elemental form, iodine may occur adsorbed on particulate material or on condensation nuclei; also, iodine is capable of forming a large number of different compounds with various materials which might find their way into gas streams either during normal plant operation or following reactor accidents. It is therefore important for experiments to be conducted to determine the forms taken by iodine under various conditions and to measure the efficiencies of iodine collectors for these forms. (W. E. Browning)

Safety in Radioisotope Source Fabrication

Radioactive sources are used widely in industry,[16,17] medicine,[18] and agriculture[19] for many applications. Sources are used for radiography,[20,21] teletherapy,[22] sterilization,[23] gauging,[24] analysis,[25] initiation of physical and chemical changes in matter,[26,27] dissipation of static electric charges,[28] and the production of heat or electricity.[29] Of the total quantities of radioisotopes sold in the United States, much more is incorporated into sources than is produced in solution form. During the year ending June 30, 1960, ORNL sold 289,000 curies of radioisotopes, either as sources or as material to be fabricated into sources, but sold only 25,000 curies of radioisotopes in solution form. A steadily increasing demand for radioactive sources is forecast to extend to megacurie quantities within the next 5 to 10 years for food processing and for special electrical generating units. Because of the present and forecast general usage of radioactive sources, a fabrication technology resulting in a safe, dependable source is of vital concern to the AEC and the general public.

A radioactive source may be defined as an encapsulated radionuclide which emits either beta, gamma, alpha, or neutron radiation. A solid radioactive substance, either a metal or a compound, is usually incorporated into the source. However, sources may be made by using a gas, such as Kr^{85}, or by using a gas sorbed in a metal, as in a tritium-zirconium source. Also, a technique for entrapping a radioactive gas in a clathrate compound for use in a source has been reported.[30]

The principles of source fabrication involve the selection of the source compound, the encapsulating material, the thickness of the capsule, the sealing method, and the inspection procedure.[31] In all these considerations the maximum utilization of the emitted radiation must be balanced against the necessity of producing the safest, most dependable source for the customer's use.

The desirable characteristics of a source compound are radiation stability, inertness

toward the source capsule, high melting point, low vapor pressure, low solubility, and maximum concentration of radioactivity. Each source compound is chosen to include these characteristics to the extent possible based on the required chemical-processing and source-fabrication steps. Some examples of source compounds now used at ORNL are listed in Table IV-1.

Table IV-1 EXAMPLES OF SOURCE COMPOUNDS USED AT ORNL

Radionuclide	Source compound	Type of radiation
Co^{60}	Co metal	Gamma
Cs^{137}	CsCl	Gamma
Ir^{192}	Ir metal	Gamma
Ce^{144}	CeO_2	Beta, gamma
Sr^{90}	$SrTiO_3$	Beta, bremmstrahlung
Pm^{147}	Pm_2O_3	Beta
H^3	H^3Zr	Neutron production in particle accelerators
Fe^{55}	Fe metal	Gamma
Sb^{124}-Be	Sb-Be metal	Neutron

Sources of $Cs^{137}Cl$ are prepared[32] by pressing pellets of the dried powder at 20,000 psi. Sources of CeO_2, $SrTiO_3$, and Pm_2O_3 are also prepared by pressing pellets. The compounds are obtained by calcining the oxalates of the elements. After the pressing step, the pellets are sintered at 1400 to 1550°C in order to produce a dense, ceramic material.

The selection of the material of the capsule for enclosing the radioactive material is made on the basis of resistance to corrosion by atmospheres normally found in laboratories, resistance to damage from temperature changes, durability, and workability. An austenitic stainless steel is normally used as the capsule material by ORNL. However, capsules are furnished to the customer's specifications and have included materials such as Hastelloy, Inconel, and platinum for special uses. A standard teletherapy capsule is supplied by commercial source fabricators in which the inner stainless-steel capsule is enclosed in a tungsten alloy capsule having standard machined threads and a window for inclusion in teletherapy machines.

The thickness of the capsule material must be chosen to balance the maximum transmission of the emitted radiation against the desired strength and integrity of the capsule. The standard capsule for Cs^{137} sources prepared by ORNL is composed of two concentric cylinders, each of which has a wall thickness of 0.040 in. The transmission of the Cs^{137} gamma ray through a 0.040-in. thickness of stainless steel is 92 per cent. However, for beta sources, thin stainless-steel foil must be used in order to obtain an appreciable fraction of the beta emissions. Extreme care must be taken in the design and fabrication of thin-foil source closures to guard against melting or puncturing of the foil during welding or handling. It has been found necessary to inspect each piece of foil for minute pinholes by a dye penetrant method prior to its use in a source capsule.

The radioactive material is usually enclosed within two, concentric, welded capsules. This double encapsulation provides an additional factor of safety in case a leak should develop in one of the capsules. Metallic pieces of Co^{60} are also nickel plated in order to minimize the spread of contamination during handling and to reduce corrosion. This plating does not, however, prevent surface contamination from the Co^{60} which diffuses through the nickel film.

All capsules supplied by ORNL are now sealed by inert-gas-shielded nonconsumable-electrode welding, with no filler metal added.[33] Formerly, brazed, soldered, or gasketed seals were supplied, but they were found to be subject to leakage because of cracking and corrosion at high temperatures. Sources have been inadvertently subjected to high temperatures when they were insulated without due regard for the radiation heating generated within the source. If all the beta radiation is absorbed in the source and practically all the gamma radiation is absorbed in the shield, the radiation heating from a 1000-curie source of the specified isotope is as follows:

Radioisotope	Beta heating, watts	Gamma heating, watts	Total heating, watts
Co^{60}	0.6	14.7	15.3
Ce^{144}-Pr^{144}	7.1	0.4	7.5
Sr^{90}-Y^{90}	6.3		6.3
Cs^{137}-Ba^{137}	1.1	3.6	4.7
Pm^{147}	0.4		0.4

After encapsulation a source is inspected for leaks by immersing it in water heated nearly to the boiling point. The presence of a leak is determined by the presence of bubbles, which

would be forced through any hole by the hot pressurized air within the capsule. In special cases a helium leak-detection test or an even more sensitive Kr^{85} leak-detection test can be applied. After the capsule passes the leak-detection test, it is checked for contamination by smear techniques. If it is found to be free of contamination (less than 200 beta-gamma counts/min), it is stored for seven days and the inspection steps are repeated. After it has passed the second inspection, the source is approved for shipment. Sources held in storage more than one week are retested prior to shipment.

Source fabrication operations are usually performed by remote control in shielded cells equipped with master-slave manipulators and a viewing window filled with lead glass.[34] Prior to the assembly of the source in the capsule, the desired source compound is produced by chemical processing and is assayed for radioactivity concentration and radioactive and stable impurities. Preirradiated Co^{60} pieces are cleansed of surface contamination prior to encapsulation, but iridium pellets are encapsulated in 2S aluminum pellets before irradiation in a reactor. The Fe^{55} sources are prepared by fusing irradiated elemental iron to a platinum disk.[35] The tritium-zirconium targets are fabricated by depositing evaporated zirconium on a platinum metal backing and then impregnating the zirconium with tritium.[36]

In the case of CsCl, CeO_2, Pm_2O_3, and $SrTiO_3$ sources, pellets are formed under high pressure in a hydraulic press. Pellets of standard sizes are available, but pellets of other sizes can be furnished to the customer's specifications. The pellets of CeO_2, Pm_2O_3, and $SrTiO_3$ are sintered by heating in a high-temperature furnace at about 1500°C for 24 hr. As a result of pellet shrinkage during the sintering process, the dimensions of the "green" pellet must be determined carefully in order to achieve the specified dimensions of the sintered pellet. Such ceramic pellets have the desirable characteristics of being extremely durable, corrosion resistant, and heat resistant.

After the desired form of the radioisotope is obtained, it is inserted into the capsule by the use of the master-slave manipulators, and the capsule is sealed by welding. The inspection procedure is then performed, as previously described.

ORNL is the major AEC supplier of encapsulated radioisotopes. Other AEC suppliers are the Mound Laboratory, which supplies polonium alpha sources and polonium-beryllium neutron sources, and the Brookhaven National Laboratory. Fabrication services are also available from commercial manufacturers for some sources. These manufacturers are listed in the Isotopes Index published by Scientific Equipment Co., Indianapolis 19, Ind. Such listings can also be obtained from the AEC Office of Isotope Development, Washington 25, D. C.

(E. Lamb)

Radiation-Safety Improvements to ORNL Facilities

Since January 1960, ORNL has been engaged in an extensive program of facility improvement. In scope it involves 28 projects and will entail the expenditure of approximately $3 million. As of November 1, six projects had been completed, and it is expected that the entire program will be finished by July 1961.

This improvement program was largely inspired by three incidents that took place late in 1959. On Oct. 29, 1959, a leak occurred in an evaporator steam coil with a resultant release of 2000 curies of radioactivity to the Laboratory's process waste system, which normally contains only low-level contamination.[37] This experience clearly demonstrated that liquid-waste-system improvements were needed, both for normal operation and for coping with emergencies. The second incident occurred on Nov. 11, 1959, when ruthenium, which had deposited on rust in ducts at the base of a stack used for gaseous-waste discharge, was dislodged during maintenance operations.[37] It is estimated that about 15 curies of ruthenium was released. Low-level contamination of portions of the Laboratory in the neighborhood of the stack resulted from this release of radioactivity. This occurrence indicated that improvements in the gaseous-waste-handling system were also needed. The final incident, which occurred on Nov. 20, 1959, involved an explosion in a radiochemical plant evaporator.[38] The explosion opened the cell door, and about 600 mg of plutonium was expelled to the Laboratory environs. This incident showed the desirability of modifying each facility so that it would contain its hazardous contents in the event of an accident.

On Dec. 11, 1959, by an ORNL management dictum, all radioactive operations involving large quantities of radioactivity were suspended until the following specifications could be met:

1. Each facility must be capable of confining the activity handled so that, in the event of the maximum credible accident, the health of personnel and the continuity of other Laboratory programs are not jeopardized.

2. Two lines of defense must be provided to prevent the escape of radioactive materials from operations via gaseous- and liquid-waste streams.

In the following discussion some of the essential features of the ensuing facility-improvement program are presented. Although they are not discussed here, procedural and operational controls have also been instituted, and these are felt to be parallel in importance to facility safety.

Containment of Radioactive Operations

In general, the facility changes for containing the maximum credible accident entail modification to provide containment of the type used for the Oak Ridge Research Reactor (ORR).[39] The ORR provides the following series of containers to prevent radioactivity release: (1) the fuel-element cladding, (2) the reactor vessel and pool water, and (3) the reactor building, which is placed under a 0.3-in. H_2O vacuum in the event of an emergency and the exhaust air is passed through a caustic scrubber before discharge to a stack.

The Homogeneous Reactor Test (HRT) type of containment was also considered.[40] To prevent radioactivity release at the HRT facility, the reactor vessel and associated equipment are constructed to rigorous specifications, and the reactor cell is constructed to leaktightness specifications which are such that the quantity of activity leaving the cell will not exceed an acceptable amount in the event of the maximum credible accident. As may be seen, the HRT type of containment is premised upon a very low cell leakage rate, namely, 10 liters/min at 15 psig. In the present equipment-modification program, it has not proved practical to achieve this degree of leaktightness in chemical plant cells or in hot cells. Therefore the ORR type of containment is being strived for.

Chemical Plant Containment. Chemical plants usually consist of processing equipment housed in cells within a building. The following criteria are used as guides for modifying such facilities to meet the Laboratory's containment specifications. A vacuum equal to or greater than 1 in. H_2O is established to assure a positive flow of air from the building into the cell. Cell exhaust capacity equal to or greater than 0.1 cell volume per minute is provided to assure the absence of an explosion hazard in cells handling flammable solvents. It also assures that, in the event of an explosion which pressurizes the cell, the cell will be returned to the negative 1-in. H_2O pressure in a short time. A leakage rate equal to or less than 0.01 cell volume per minute at a pressure differential of 2 in. H_2O is specified to assure that contamination escaping the cell will be at a minimum in the event of cell pressurization. Seals are designed to withstand a pressure of 10 in. H_2O or more to assure that the seals at places such as cell-block covers will not be destroyed in the event of an explosion. In the design of new facilities, shielding will be provided which will withstand the pressure produced by the maximum credible accident. No attempt is made to meet this objective in existing facilities.

Under an emergency condition the containment-building vacuum will be 0.3 in. H_2O or greater. This value was selected to balance the leeside building vacuum which is created by a 30-mph wind. In the Oak Ridge area the wind velocity is less than 30 mph 99.97 per cent of the time. Under an emergency condition the containment-building air will be evacuated through the cell ventilation system, which is provided with roughing and absolute filters. The building leakage rate is to be equal to or less than 6×10^{-3} building volumes per minute. This value will assure that radioactive release from the building will be at an acceptable minimum. The time required to bring the building to a negative pressure of 0.3 in. H_2O is to be equal to or less than 20 sec. In this time the radioactivity escaping the building, because of a leeside negative pressure differential, will be an acceptable minimum.

If an ORNL facility meets all these criteria, it will not present a hazard to personnel outside the building or to the continuity of neighboring operations. A mathematical expression, which takes into consideration all the criteria, has been developed to permit assessment of the hazard that any operation offers its neighbor. In this expression, flexibility in meeting specific

elements of the criteria is provided; for example, failure to meet the building leaktightness specification may be compensated for by attaining the desired building negative pressure more quickly than the required 20 sec. Final approval for operation of a facility following modification rests upon the over-all mathematical assessment of its safety based on measured values of containment criteria.

Hot-Cell Containment. Tentative containment criteria have been set up for hot cells to be used for handling large quantities of radioactive materials. Although the specifications are not yet definite, the following elements that are felt to be important are being used as a basis for modifying existing facilities:

1. A minimum vacuum of 1 in. H_2O must be maintained, and there must be a minimum flow of 100 ft/min through all openings.
2. An alarm must be provided to indicate the loss of cell negative pressure.
3. Exhaust air from hot cells must pass through at least one roughing filter and an absolute filter and be continuously monitored. A suitable alarm must be provided to indicate the presence of radioactivity.
4. Operations producing large quantities of airborne contamination must be conducted in a second enclosure within the cell. This enclosure must be provided with an exhaust that is filtered before discharge into the cell ventilation system.
5. Chemical contaminants that are capable of damaging the cell exhaust filters must be removed from the cell ventilation air.

Operations in Laboratories

Those laboratories which handle sizable quantities of radioactivity, i.e., greater than 1 g of plutonium or equivalent, must be capable of containing the hazardous material in the event of the maximum credible accident. Containment is to be achieved by requiring a minimum negative pressure of 1 in. H_2O in glove boxes and equipping them with alarms to indicate the loss of negative pressure. Containment of hazardous material within the laboratory room or building is to be assured by providing positive air flow from cold areas of the building to hot areas, and the laboratory ventilation air is to be treated by roughing and absolute filters prior to discharge.

Gaseous-Waste-System Modifications

Before being exhausted to the atmosphere, all dissolver and processing vessel off-gases must be treated for the removal of constituents which would be detrimental to filter media, gaseous fission products (exclusive of the rare gases), and particulate material. The treatment employed varies from facility to facility, but, in general, nitric acid scrubbing is employed to remove basic constituents, such as ammonia, and sodium hydroxide is employed to remove volatile ruthenium compounds and to remove acidic components, such as nitric acid vapors. The dissolver and processing vessel off-gases are then scrubbed with water and heated to prevent condensation of water vapor. Finally, particulate material is removed by passing the gases through roughing and absolute filters. In addition to the treatment given off-gases at each facility, a secondary cleanup is provided by a central off-gas treatment facility that is capable of handling gas at a rate of 4000 cfm.

Suitable monitors are being installed in all branch ducts leading to stacks to permit prompt identification of the source of gaseous waste contamination and to detect failure of the primary waste treatment at the facility. Gas monitors are also being installed in the stacks to permit maintaining an inventory of radioactive discharges. The stack monitor consists of a step paper tape for the collection of particulate material and an ambient-temperature charcoal bed for sorption and monitoring of fission products.

Liquid-Waste-System Modifications

Major improvements are being made in the liquid-waste system to preclude the unexpected release of radioactivity to the environment and to provide for lowering the amounts of activity discharged over the longer term. The bulk of the environmental contamination stemming from ORNL operations occurs by way of the process waste system, which normally contains only low levels of radioactivity.

Process wastes from the various ORNL facilities are to be continuously monitored, the results recorded, and proportional samples taken. If excessive contamination is detected, steps will be taken to determine the facility that is responsible, and it will be promptly shut down. In certain cases where materials presenting a difficult monitoring problem are

handled, holdup tanks are being provided and conventional radiochemical analyses, rather than continuous monitoring, will be employed. Further monitoring of the entire process waste stream will be performed ahead of the plant retention basin and immediately prior to discharge to the Clinch River.

Steps have been taken to provide greater holdup capacity of the process waste by increasing the storage volume of the retention basin from 700,000 to 1,000,000 gal. Means are being provided for pumping the entire Laboratory output from the retention basin through a 6-in. pipeline to a new 3,000,000-gal storage basin in the event gross contamination of the process waste stream occurred. The new basin is being constructed about $1\frac{1}{2}$ miles southwest of the Laboratory.

During the two-year period following January 1961, an expenditure of 1.7 million dollars is proposed for liquid-waste-system improvements. Storage tanks provided with cooling coils will be installed for the storage of high-level wastes. These wastes will be evaporated to a minimum volume at each facility prior to routing to the tanks. Intermediate-level wastes will be composited, evaporated, and stored in noncooled tanks. Radioactive contamination of the low-level waste which enters mainly from reactor and fuel-storage canals will be minimized by treating the canal water by filtration and ion exchange. Additional reductions in the radioactive content of the low-level waste will be achieved by improvement of the existing waste-treatment plant.

Conclusions

This extensive program for improving the radiation safety of ORNL facilities is necessary primarily because of changes in research activities. Higher activity levels are being handled, the research and development work being conducted has become more diversified, and the number of radioisotopes and heavy elements being processed has increased. The planned facility changes will assure safe handling of the quantity and variety of hazardous materials that are visualized for future programs. The over-all cost of the modifications is relatively small, i.e., less than 4 per cent of the total investment in facilities at ORNL, and less than 10 per cent of the value of the individual facilities being modified. The experience with radiation incidents during the past year has left the conviction that all activities involving significant quantities of radioactive materials should be provided with a measure of safety that is comparable to that being achieved in this program. (F. R. Bruce)

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Section V

CONSEQUENCES OF ACTIVITY RELEASE

Reexamination of the Sr^{90} Problem

Great interest in the toxicity of Sr^{90} has been stimulated by international concern over fallout from nuclear tests, the possibility of area contamination during a major reactor accident or a smaller, but more probable, laboratory accident, and the use of Sr^{90} in biological experiments. Although the research effort in this field is active (140 publications are listed in *Nuclear Science Abstracts*[1] for the first six months of 1960), many problems remain unsolved. Nevertheless, recent contributions to the literature have provided a better general understanding of the problem and some specific reassurance with respect to fallout.

Strontium-90 is one of the most hazardous of the bone-seeking fission products. It has long physical, biological, and effective half lives (28, 50, and 18 years, respectively), and it and its daughter product Y^{90} are beta emitters with an effective energy of 5.5 Mev.[2] In soluble form Sr^{90} is absorbed by plants through the leaves and from the soil, and, when eaten (or inhaled) by man or experimental animals, it follows the same general metabolic pathways as calcium. Fortunately the body discriminates against strontium in the transfer of these materials from diet to bone. The now widely accepted calcium-strontium discrimination factor is 4. Since Sr^{90} follows calcium, it is readily transmitted to human and cow's milk and to the developing fetus, and it is deposited and retained in the skeleton.

Bishop et al.,[3] who studied the retention and excretion of Sr^{89} in two normal male adults who were given 0.5 μc of Sr^{89} intravenously, report that strontium is highly mobile during the first two days after injection. The authors believe that about 65 per cent of the injected dose is probably deposited in the skeleton within the first 10 to 12 hr. Apparently, however, this initial deposition is freely exchanged with the calcium of the extracellular fluids and is rapidly removed. Seventy per cent of the injected dose (70 per cent of this in the urine) is excreted within the first 20 to 30 days. During a second stage that starts after the first two days, strontium is more closely bound to the apatite crystals of the calcified tissue, and exchange proceeds at a much slower rate. About 15 per cent of the original dose is excreted, about equally in urine and feces, with a half-period of 50 days. During the final stage, strontium enters the lattice of the apatite crystals, where it becomes firmly bound; thereafter, exchange and excretion are extremely slow.

Strontium Body Burdens

The present maximum permissible continuous body burden[4] of 2.0 μc for Sr^{90} is based on a comparison with the known toxicity of radium. An attempt has been made to calculate the amount of Sr^{90} which would deliver the same relative biological effectiveness (RBE) dose when deposited in the bone as the maximum permissible concentration (MPC) (0.1 μc) of Ra^{226}. Although elaborate calculations have been performed and a useful MPC level has been determined, this approach has serious shortcomings.[5] The concentration of mineral elements in bone is uneven, at best, and depends not only on the age of the individual but also on the biological activity of that portion of the bone. The radiation dose at any point in the bone depends on the concentration of the isotope at that point only if the linear dimensions of the deposit are comparable to the range of the radiation emitted. The maximum range in bone of the beta particles from Sr^{90} and its equilibrium daughter Y^{90} is about 0.6 cm, whereas the maximum range

of the alpha particles from Ra^{226} is only approximately 35 μ. It is also necessary to know what portions of the soft tissue, i.e., osteocytes (bone cells), Haversian canals, or marrow spaces, are most vulnerable to the radiation. The distances of the tissues from the radioactive deposit, as well as the volumes of the critical soft tissues being irradiated, are important. It is apparent that more information is needed on the actual amount of Sr^{90} which, when deposited in the bone, will cause damage comparable to that observed from known deposits of Ra^{226}. Long-term experiments that should provide some of the desired data are presently under way at the University of Utah.[6]

Engström and coworkers[7] have expressed concern over the nonuniform distribution of Sr^{90}. It is deposited more in areas of maximum new bone growth, called reactive sites, than in established bone structure. Although this process is most intense in young, growing individuals, there is some constant activity in adults. The remodeling of the microscopic structure in the long bones during the slow but continuous process of resorption and deposition results in a number of microscopic "hot spots." "From a health hazard point of view, therefore, the critical parts of the osseous tissue receiving the largest doses of internal radiation are mainly located in the spongious bone tissue as it has the highest isotope uptake. It is in this type of bone that the blood-forming bone marrow is located, a fact which enhances the risks of radiation damage."[7]

On the other hand, Hindmarsh et al.[5] have shown that, although radium and strontium have essentially the same distribution in bone, the nonuniformity of the radiation dosage for Ra^{226} is greater than that for Sr^{90}. They have calculated a "dosage nonuniformity factor" that is the ratio of the radiation dose in the hot spot to the average dose, assuming a uniform distribution throughout the skeleton. According to their calculations, the nonuniformity factor varies from 1 to 16 for Ra^{226}, whereas it varies only from 0.9 to 3.1 for Sr^{90}. These calculations suggest that the hot-spot problem may not be as important as Engström believes.

The beta-particle energy of Sr^{90} which is absorbed by the surrounding tissue may cause many complex biochemical changes. These assume real significance when they lead to injury or illness that causes temporary or permanent disability of any part of the body. The two effects, which quite naturally have received most of the attention thus far, are malignant degeneration of bone (osteogenic sarcoma) and malignancy of the blood-forming tissue located in the bone marrow (leukemia).

Finkel[8] at Argonne National Laboratory reported that, at dosages of 1.3 through 88 μc per kilogram of body weight, treated mice, on the average, died a little sooner than the control animals, but their deaths "were not obviously associated with any particular disease." At dosages of from 200 through 2200 μc per kilogram of body weight, the primary cause of death was neoplastic disease. A number of investigators have confirmed the carcinogenic effect of both Sr^{89} and Sr^{90} and their bone-seeking daughter products in other animals.[9,10] Other neoplasms that were similar to human leukemia occurred in mice examined by Finkel, although the total incidence did not appear to be greatly influenced by the dose. The neoplasms did, however, appear much earlier in the animals that received the larger doses (i.e., 88 μc/kg or more). Thus Sr^{90} is apparently capable of causing bone cancers and leukemia in experimental animals that have been exposed to relatively large doses.

Strontium Fallout

Before considering the likelihood of producing bone cancers and leukemia in humans who have been exposed chronically to low doses, it is useful to review the present status of the fallout problem. Fallout is presently the greatest source of human contamination with Sr^{90}. According to official estimates,[11] a total fission yield of 92 megatons or 9.2×10^6 curies of Sr^{90} had been released into the atmosphere by November 1958. Four megacuries had come down in local fallout, 3 Mc had been deposited world wide, and approximately 2.2 Mc remained in the stratosphere. The amount still remaining in the stratosphere, assuming no more material is injected, is now believed to be considerably less than originally estimated. At one time, before the Russian test series in 1958, it was feared[12] that 2.5 to 3.2 Mc was still in the stratosphere and would reside there for many years. Soviet scientists[13] on the United Nations Scientific Committee on the Effects of Atomic Radiation took exception to the 1958 American assumption that stratospheric equilibrium had already been reached and that future fallout would remain

relatively constant. The Soviet delegation believed that equilibrium might not be reached for 10 to 30 years and that contamination would be 2 to 2.5 times the first assumption.

Fortunately, a much shorter stratospheric residence time has been established by the high-altitude sampling program of the Defense Atomic Support Agency. Feely[14] estimates that the mean stratospheric residence half-time is less than one year. The British and American injections, which were made into the tropical stratosphere, have an appreciably longer residence time than the Russian injections, which were made into the polar stratosphere above Siberia. The total stratospheric inventory of Sr^{90} up to the time of the last USSR test series in July to November 1958 was estimated to be 1 Mc, and these tests apparently added 1.5 Mc to bring the total to 2.2 Mc. The residence time of the Sr^{90} injected during the latest tests by the USSR was probably quite short, i.e., on the order of four to nine months. Thus, most of the Sr^{90} in the stratosphere had "fallen out" by 1960.

Bone Analyses

The results of bone analyses are more useful in evaluating the health hazard than fallout measurements, since they give evidence of the amount of Sr^{90} that is actually absorbed in human bone. The average concentration of Sr^{90} in the skeleton for most of the world population at the end of 1956 was reported[12] to be about 0.20 $\mu\mu$c per gram of calcium. It was predicted that, if there were no further tests after mid-1957, the Sr^{90} concentration in children in the northeastern United States would reach a peak level of 2.9 $\mu\mu$c per gram of calcium, and, if testing continued at the 1956–1957 rate, the equilibrium bone level for the population might reach 21 $\mu\mu$c per gram of calcium. A more recent report[15] from the same laboratory indicates that the previously predicted levels were much too high. This was due to overestimates of the size of the stratospheric reservoir and an underestimate of the rate-of-fallout factor. The equilibrium level will probably be 5 to 10 times lower than that predicted. The concentration in new bone will reach a maximum in 1960.

Pathological Effects

The seriousness of the health hazard of Sr^{90} would seem to depend primarily on the pathological effects of continued low-level exposures to the skeleton and bone marrow. In 1957, Lewis[16] concluded that the incidence of Sr^{90}-induced leukemia in a stationary population maintaining a constant level of 0.1 μc per kilogram of calcium in their skeletons would be three to six cases per million persons per year. The probability of developing radiation-induced leukemia for the "average" individual was 2×10^{-6} per rad per year. Lewis assumed that there was a linear dose relation even down to low dose levels and that there was probably no threshold below which radiation was not leukemogenic. Lewis's estimates of the hazard have been widely quoted in both the lay and scientific press. It is an understatement to say that he has created a great deal of world-wide apprehension.

There appears to be little evidence to support Lewis's original hypothesis. Brues,[17] Lamerton,[18] and Cronkite et al.[19] have concluded that, at the present time, there is insufficient information to form any conclusion concerning the presence or absence of a threshold dose, or for assuming any particular relation between the incidence of tumor and dose. With an even more pointed comment, Glucksmann et al.[20] say that "none of the animal experiments have indicated a linear relationship between tumor incidence and dose."

There are a number of nonlinear responses in radiation-induced tumors. For example, lymphoid tumors induced in mice by total-body X ray increase by more than tenfold as the X-ray dose is increased only three times.[17] Lack of a dose-rate dependence has been used as crucial evidence of the linearity of genetic point mutation. If somatic mutation is important in carcinogenesis, then dose rate should *not* be important. Only the total dose would be significant, regardless of the length of time over which it was given. However, in experimental cancer there is usually a *considerable* dose-rate dependence. As a rule, the number of cases for a given total dose drops off with lengthening of the total exposure time.

Upton et al.[21] have reported on the results of total life-span observations on a large number of mice that had been exposed to varying amounts of radiation during a test detonation. The authors concluded that there was no consistent relation between the frequency of malignancies and the radiation dose. The incidence of tumors in some animals rose progressively with increasing amounts of radiation, whereas in some other

animals it actually decreased. In other cases the peak incidence occurred at intermediate dose levels. "In no instance did the observed tumor incidence vary as a simple, linear function of the radiation dose."[21]

The strict linear relation that Lewis[16] proposed almost requires some form of somatic mutation to explain it. It is now quite apparent that cancer is caused by a great many factors in a complex relation. As Mole has said,[22] "It is probably too naive to expect a linear dose-response relation for carcinogenesis by radiation even if the mutation hypothesis were true, for radiation is known to kill cells, and mutated cells will be no exception." The apparent influence of dose rate on the incidence of thymic leukemia in mice amounts to a formal disproof of the point mutation hypothesis.

Regarding leukemia alone, one is forced to conclude, with Brues,[17] that "present data on human leukemogenesis by radiation fail to indicate a linear relation between dose and effect" at low levels. Hollingsworth,[23] however, has reported that the Hiroshima data support a linear relation, but only to acute single doses above 75 r. It appears that radiation is leukemogenic at high dose levels, but there is little or no evidence that it is leukemogenic at low levels, especially with chronic exposure.

Conclusions

Although Sr^{90} may still be one of the most hazardous of the bone-seeking radioisotopes, present evidence of its toxicity does not support action which would seriously jeopardize the orderly and careful development of the reactor industry. The most important unsolved problem is the effect of chronic low-level exposure. This is tremendously difficult because of the "prohibitively large numbers of animals [which] would be required to obtain significant experimental data at the low levels of incidence of late effects."[24] Long-term, chronic, low-level exposure studies in larger mammals, which are under way, should, however, provide much needed information on the carcinogenic hazard of this material.

The present 2-μc MPC body burden of Sr^{90} probably has a wide margin of safety. Nevertheless, since there is still too little known about the biological effects of radiation and since such excellent means of detection are available, there is little excuse for allowing unnecessary human exposure.

(T. A. Lincoln, M. D.)

Accident Dosimetry

In addition to the hazards associated with the ingestion of radionuclides, hazards are also encountered from external radiation fields. There have been seven major criticality accidents within the past 15 years. Three of these were in 1958: the Y-12 accident[25–29] at Oak Ridge on June 16, 1958, the Boris Kidrich accident[30] in Yugoslavia on October 15, 1958, and the Los Alamos accident[31] on December 30, 1958. These three accidents resulted in serious exposure of 15 persons, two of whom died. None of the exposed persons in any of the accidents were sufficently monitored so that their exposure could be readily determined. Consequently, groups involved in radiation dosimetry research were requested by the AEC Division of Biology and Medicine to provide guidance and instrumentation for organizations requiring accident dosimetry systems. In a report edited by Hurst and Ritchie,[32] a comprehensive dosimetry system is outlined which utilizes the most widely accepted techniques. The complete dosimetry system includes fixed-detector stations for monitoring neutrons and gamma rays in areas where criticality is possible; personnel meters, including a device for determining the orientation of the wearer relative to the source; and analyses of blood samples for Na^{24}.

Area Monitors

Threshold detectors[33,34] for neutrons and anhydrous chemical[35] or metaphosphate glass[36] detectors for gamma rays are recommended for complete fixed-station monitors. Sulfur, gold, gold in cadmium covers, and glass or chemical detectors should also be used liberally as secondary fixed stations. Unfortunately, the cost of a complete fission-foil detector limits its use to minimum coverage. With a complete threshold detector station, the neutron flux above each of the following energy thresholds is measured: 4 kev (Pu^{239} in 2.2 g of B^{10} per square centimeter), 0.7 Mev (Np^{237}), 1.5 Mev (U^{238}), and 2.5 Mev (S). Gold and gold shielded by cadmium are used as the thermal-neutron (2200 meters/sec) detectors.

Personal Meters

Several personnel monitoring badges have been described. The most complete is the ORNL type,[32] which includes film with several filters, glass, sulfur, gold and gold plus cadmium, indium foil, and, in some instances, a chemical detector that is sensitive to both neutrons and gamma rays. The indium foil is included because it serves as a sensitive indicator of gross neutron exposure. By "scanning" each person's badge with a Geiger-Mueller (G-M) tube or a scintillation counter, all exposed persons can be detected quickly. Because the half life of In^{116} is only 54 min, readings must be made within a few hours. The chemical detector (a hydrocarbon), by visible color change, serves for the detection of both neutrons and gamma rays. It exhibits a distinct color change upon exposure to approximately 30 r of gamma rays, 10 r of 100-kev X rays, 40 rads of fast neutrons, or 5 rads of thermal neutrons.

Blood-Sodium Analyses

Sodium-24 analyses played an important role in the dose determinations for persons involved in the three criticality accidents of 1958. A report[37] of a detailed experimental study of the accident that occurred in Yugoslavia includes information on the blood-sodium analyses and a description of an experiment which demonstrated that a homogeneous sodium solution in shell phantoms is a suitable substitute for a living animal for sodium-activation studies. A burro was submerged in a tank containing aqueous NaCl solution and exposed to fast neutrons from the Godiva II reactor. The tranquilized burro was bound with his legs folded close to his body so that the tank could be kept small. In a second exposure the NaCl solution volume was the same as the volume of solution and burro in the first exposure. Thus the space initially occupied by the burro was filled with a homogeneous solution with the exact shape of the burro. A weighted average of the Na^{24}-to-Na^{23} ratio in the burro's blood and in the solution from the first exposure compared within ±5 per cent with the ratio for the second exposure.

In general, Na^{24} can be determined by gamma-ray counting of blood samples drawn from exposed persons. The blood can be used as whole blood with an anticoagulant added, or the serum may be drawn off after clotting and centrifugation. *In vivo* counting has also been used, and research is under way at several installations to improve these techniques.

Cofield[38] has studied the use of an *in vivo* gamma spectrometer for sodium-activation analyses. An apparent lower limit of detection for fast neutrons from an assembly such as the one involved in the Y-12 accident is of the order of 10 mrads for measurements made within one half life of Na^{24}. However, the neutron-energy dependence of the Na^{24}-to-Na^{23} ratio must be kept in mind; for a softer spectrum[37] the Na^{24}-to-Na^{23} ratio is significantly higher per tissue rad of fast neutrons, and for a harder spectrum the ratio would be slightly lower for a given neutron dose. Scintillation counters are generally best for all methods of Na^{24} counting because of their sensitivity and suitability to spectral analysis or differential counting between two energy levels.

Tissue Activation

Neutron activation of tissues other than blood is being studied by several groups in the United States and abroad. A report by Rovner[39] describes a facility for studying the activation of elements in tissue and lists some preliminary results for black mice. The machine used was a Cockcroft-Walton type of accelerator for producing the ${}_1H^3({}_1H^2,n)_2He^4$ and ${}_1H^2({}_1H^2,n)_2He^3$ reactions. An exposure of about 70 rads yielded a measurable urine activation. The counting methods were not described.

Radiation Alarms

Research and development of radiation alarm systems continue at several laboratories. Perry and Maddock[40] have reported a relatively simple alarm unit that gives visual and audible warning when the gamma radiation reaches a predetermined dose rate. Three levels of dose rate are indicated by lights: less than 30 mr/hr, 30 to 300 mr/hr, and greater than 300 mr/hr. As in most such systems, an ionization chamber and a linear current amplifier are used. The alarm-triggering system utilizes two phototransistors in conjunction with the microammeter and three lamps. The lamps used are too susceptible to failure for this application, but the authors indicate that this is to be remedied.

Neutron-Detector Developments

Many groups are studying neutron detectors that may have application in nuclear-accident monitoring. However, most of the literature concerns only thermal detectors or thermal detector-moderator combinations for fast neutrons. These detector-moderator assemblies are generally insensitive to fast-neutron energy and are, in principle, similar to the Hanson-McKibben long counter.[41] Typical of this type of detector is the cobalt-paraffin assembly reported by Smith.[42] A cobalt disk embedded in a cadmium-encased paraffin moderator can be removed at intervals for analysis with a gamma-ray scintillation spectrometer. Because of the 5.3-year half life of the Co^{60} resulting from the $Co^{59}(n,\gamma)Co^{60}$ reaction, neutron fluxes may be integrated for relatively long periods. An integrated flux of 1×10^7 neutrons/cm^2 may be measured with the system, and the flux intensity may be a small fraction of tolerance levels. For a known neutron spectrum, such detectors have application in accident dosimetry.

(J. A. Auxier)

Some Recent Concepts in Nuclear Waste Disposal

A growing awareness of the problem of disposing of radioactive wastes is manifested by the ever-increasing number of publications on this problem. Recent papers have been devoted to a review of the waste problems associated with a nuclear power economy, an analysis of the possibility of using fission products as heat sources, and descriptions of the management of low- and intermediate-activity wastes at several existing foreign and domestic atomic energy sites.

In a general review of the waste-disposal problem, Koltzbach[43] points out that the overall problem can be broken into two parts: (1) the management and disposal of materials that are greatly diluted and widely dispersed geographically and (2) the almost perpetual containment of large quantities of high-activity waste from spent-fuel processing. The release of low-activity wastes can have a deleterious effect in such nonnuclear enterprises as the photographic film, pharmaceutical, and food-processing industries. The control of this hazard will be difficult because of the number of sources and people involved, but the means for control exists since the total quantity of radioactivity issued can be monitored. Tank storage of high-activity waste currently costs as much as $2 to $3 per gallon and suffices only as an interim measure because the tank lifetimes are relatively brief. Possibilities currently seen for permanent disposal of high-activity wastes include conversion of the liquids into solid products that can be shipped and stored in mines or vaults and direct disposal of the liquids into cavities in salt. Costs can be expected to exceed those of tank storage, and nuclear-chemical-plant site selections should be made with a consideration of the waste-transportation and -disposal problems in mind.

Hardin and Beckelheimer[44] have concluded that reducing the cost of waste disposal through the use of fission products as large-scale heat sources is not feasible. In exploring this possibility, the cost of producing a minimum of 10^6 Btu/hr over a 28-year period in a fission-product "furnace" was estimated and compared with the current market value of the heat, taken as $0.60 per million British thermal units. The cost of producing the heat from 100-day-decayed fission-product mixtures was computed from a consideration of capital investment, waste shipping, and operating and maintenance costs, and it amounted to between $464,000 and $568,000; whereas the value of the heat was only about $250,000. Although the conclusion drawn from this study is not unexpected by those who have had occasion to consider the feasibility of recovering fission-product heat for useful purposes, it is remarkable that the issue remained so clear-cut after an optimistic value of only $100,000 was taken as the cost of the fission-product furnace. It is this reviewer's opinion that the actual cost of any such furnace could well be a factor of 10 greater.

The treatment and disposal of 300 to 350 m^3 of low-level liquid wastes per day at Marcoule have been described by Galley and Cantel.[45] These wastes, which contain activity in the range of 1 to 50 mc/m^3, arise in the later stages of the plutonium separations process, as well as from cell and equipment decontamination washes, laboratory drains, and process cooling water. They are collected in stainless steel or acid brick-lined concrete tanks and treated either by a calcium phosphate or a strontium phosphate–nickel ferrocyanide scavenging precipitation for removal of activity prior to dilution with nonradioactive reactor cooling water

and release to the Rhone river. The sludges are recovered by using rotary filters with a diatomaceous precoat, packaged remotely in polyethylene-lined galvanized-iron drums of 200 liters capacity each, and stored temporarily in a specially prepared area pending the demonstration of suitable permanent disposal sites on land or at sea. The calcium phosphate treatment yields a gross beta decontamination factor of about 12, and, although it is efficient in the removal of Cs^{137}, it has yielded Sr^{90} decontamination factors as low as 2. The strontium phosphate–nickel ferrocyanide treatment, as applied to the higher activity wastes, yields Sr^{90} decontamination factors of greater than 30. The cost of this treatment, including sludge storage, is reported to be 2400 light francs/m^3 (~$0.02 per gallon) for the calcium phosphate treatment and 4500 light francs/m^3 (~$0.04 per gallon) for the strontium phosphate–nickel ferrocyanide treatment.

The disposal of low-activity solid wastes poses a major problem because of their large quantity and diversity. Burns,[46] in describing the British solids disposal practices, points out that these wastes cannot be diluted and released like liquid or gaseous wastes but that they must be stored with adequate shielding in a manner excluding surface and rain water. At Windscale, reinforced-concrete silos 58 ft deep, 30 ft long, and 15 ft wide, with 6-ft-thick walls, are used for solid waste storage. Below-ground concrete trenches with removable concrete lids are used at Harwell. In addition, both vertical and horizontal steel and concrete tubes with lead and concrete end plugs are employed. Direct burial of these wastes in the ground, as practiced in the United States, is used in Britain to only a very limited extent because of the greater population density and scarcity of available land, but controlled dumping of low-activity solids at sea in 100 fathoms or more of water is practiced. In all disposal operations, volume reduction is an important factor in minimizing costs. Baling of such items as paper, glassware, and thin-walled metal tubing yields a volume reduction factor of 10. Volume reduction factors of about 80 are obtained by incineration of combustibles, and scrap metal is melted *in vacuo* using high-frequency induction furnaces.

Waste-management operations at Chalk River, Canada, were described in a series of four papers prepared for the 1959 Conference on Waste Disposal at Monaco.[47] During the period September 1958 to August 1959, 2×10^5 m^3 of water containing 2000 curies of soluble beta-gamma activity was pumped into a pit 70 meters long, 37 meters wide, and 2.5 meters deep which was filled with 4- to 6-cm pebbles and allowed to seep into the sandy soil. In addition, 1.6×10^4 m^3 of chemical wastes containing 55 curies of beta-gamma activity was pumped into two other pits of the same design. Although solids have been buried in asphalt-lined cribs and trenches at Chalk River with no detection of activity leakage, cracks in the asphalt have recently been observed, and it is planned to discontinue its use in favor of concrete. Monitoring of surface water draining from the disposal area into the Ottawa River indicates that the water contains about 30 per cent of the MPC for occupational exposure, assuming all detected activity is Sr^{90}. Surveys of both the river and its associated flora and fauna are routinely carried out; and, of the species sampled, young perch have shown the greatest concentration of activity (i.e., 1×10^{-3} beta-gamma microcurie per gram of net weight, 90 per cent of which was due to P^{32}). A rugged, reliable survey instrument insensitive to cosmic radiation and capable of establishing a prestartup environmental background-activity value is described. Its use in a survey of 300 sq miles surrounding the NPD reactor site resulted in the detection of a background averaging 6 μr/hr, which is somehwat lower than the value obtained in comparable surveys made in the United States with ionization chambers.

The most recent quarterly report of waste-disposal operations at ORNL[48] indicated that, in January through March 1960, the Clinch River averaged 31 per cent of MPC_w, reputedly due mainly to the washout of Sr^{90} from the bed of former White Oak Lake. The lime-soda process water-treatment plant at ORNL treated 42 million gallons of waste during this period at a reduced efficiency of about 50 per cent due to the presence of detergents and complexing agents used in a major decontamination program. A total of 5.9 curies of Sr^{90} was discharged into the creek. A new seepage pit is being readied for intermediate-activity wastes, and attempts will be made to reduce the discharge of Ru^{106} from this and existing pits by the addition of copper compounds for fixation of this activity. During the first quarter of 1960, 860,000 gal of intermediate-activity waste containing 10,150 gross beta curies was transferred to the pits,

bringing the total quantities thus disposed to 457,000 curies in 16,123,000 gal.

The engineering details of laying two 10-in. mild-steel lines 8400 ft into the Irish Sea at Windscale during 1948–1950 are described by Day.[49] Because of the scale and nature of the operation, the work was prepared and carried out with the care and detail characteristic of the best military operations. Approximately 400-ft lengths of pipe were made up from 40- to 50-ft lengths, and these, in turn, were butt welded into three 2415-ft-long pieces and one 1135-ft-long piece. These great lengths were welded together on shore and dragged into the sea by ship. Each of the lines is rated at 60,000 gal/hr capacity under gravity flow conditions and terminates in a sea sled with diffusers for releasing low-activity effluent. Low-voltage cathodic protection is used to inhibit corrosion, and the total cost of the project is reported to be £500,000. (J. O. Blomeke)

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Section VI

CURRENT EVENTS

SL-1 Incident

An explosion occurred in the Stationary Low-Power Plant No. 1 (SL-1) reactor at the National Reactor Testing Station in Idaho at approximately 9:00 p.m. on Jan. 3, 1961, which resulted in the death of three persons, in damage to the reactor and reactor room, and in the release of gaseous fission products. At this time, it is not possible to identify completely or with certainty the causes of the incident, although the most likely immediate cause of the explosion appears to have been a nuclear excursion resulting from movement of the central control rod by the operating personnel. The explosion involved a nuclear reaction estimated to have resulted in 1.5×10^{18} fissions which, it is currently estimated, gave an integrated thermal neutron flux above the reactor of approximately 10^{10} neutrons/cm^2. Damage to the building by the explosive blast was slight; damage to the reactor core has not yet been determined. Some gaseous fission products, including radioactive iodine, escaped to the atmosphere outside the building, but the particulate fission-product material was largely confined to the reactor building.

In view of the wide interest in the SL-1 reactor incident, it has received thorough coverage by members of the press and in technical publications. Soon after the accident, the AEC appointed an investigating committee which has, to date, submitted an interim report, the essence of which is summarized above. When the final report of this investigation is available, it and all other information pertaining to this incident will be reviewed in detail in *Nuclear Safety*.

SRE Operating Experience

Editors' Note: This article was prepared by Robert W. Dickinson of Atomics International at the request of the editors of* Nuclear Safety. *It therefore represents the views of Atomics In-*

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Dickinson obtained the A.B. degree from the University of California in 1941, the M.S. degree in mechanical engineering from the U. S. Naval Postgraduate School in 1948, and the S.M. degree in nuclear engineering from the Massachusetts Institute of Technology in 1953.

During World War II he served as a naval officer aboard U. S. submarines, chiefly in an engineering capacity. He participated in seven combat war patrols, and was awarded the Silver Star medal. After obtaining the M.S. degree from the U. S. Naval Postgraduate School in 1948, Dickinson was a submarine squadron engineer with the U. S. Navy in San Diego, Calif., for two years. From 1950 to 1952, he was an assistant planning and estimating superintendent at the Mare Island Naval Shipyard in California.

Following study at Massachusetts Institute of Technology, Dickinson was connected concurrently with the Navy's Bureau of Ships as an engineer concerned with the application of nuclear propulsion for naval use and with the AEC as head of the Nuclear Components Section of the Naval Reactors Branch. He resigned from the Navy in 1956 with the rank of commander and joined Atomics International the same year.

ternational and is consequently not a critical review in the same sense as most of the material presented in this journal.

The Sodium Reactor Experiment (SRE) is a sodium-cooled graphite-moderated reactor system that operates with slightly enriched fuel at a thermal power level of 20 Mw. It was originally designed solely as a reactor experiment, but a small turbine-generator set and a steam generator were added by the Southern California Edison Company (SCEC) to dissipate the heat produced. The plant for generating electricity, including the steam generator, is owned and operated by SCEC; the reactor is owned by the AEC and is operated by Atomics International. Criticality was achieved in April 1957, and electric power was first generated on July 12, 1957. During the life of the first fuel loading, slightly more than 15×10^6 kw-hr of electricity was generated. Operation of this first core was terminated by metallurgical damage to the unalloyed uranium fuel as a result of fission-gas swelling. Coincidentally, an auxiliary coolant system failure, which will be discussed in detail below, caused restrictions of coolant flow to the fuel elements and required suspension of reactor operation on July 27, 1959. It was found that 13 of the 43 fuel assemblies had been damaged.[1,2] A new fuel loading consisting of 7.6 wt.% thorium in a uranium alloy was inserted, and criticality was achieved with the new core loading on Sept. 5, 1960.

The SRE was the first high-temperature reactor of any significant power rating to be operated in this country and, as such, has provided much information relative to sodium technology and to high-temperature reactor operation in general. A "once-through" steam-generating system is used which provides steam at 600 psi and 825° F. These conditions were set, not by the ability of the liquid-metal-cooled reactor system to generate steam at these conditions, but rather by the ability of the small 7.5-Mw(e) turbine-generator set to employ them. Summaries of SRE operating experience have been provided from time to time in periodic progress reports and, more recently, by a comprehensive report dealing with the damage to the fuel assemblies in the 1959 incident.[3] An over-all summary of the operating history to date is being prepared, and detailed descriptions of the reactor and the early operating experience have been issued.[4]

System Operating Information

The design and construction philosophy of the system emphasized the use of conventional, commercially obtainable components wherever possible. To this end, the coolant circulating pumps are simple adaptations of hot oil pumps. The stuffing box was replaced by a cooled annulus around the shaft to freeze sodium and thus seal liquid sodium in the pump casing from the supporting bearings on the shaft and the drive motors.

The primary and the secondary systems each have a 6-in. pump and a 2-in. pump, and three of these pumps have had erratic operating histories. The 2-in. pump in the primary system has bound on numerous occasions. It is normally operated at minimum speed, and minor fluctuations in the temperature of the coolant to the freeze seal result in fairly large variations in the torque required to shear the frozen-sodium shaft seal. These fluctuations have been reduced as experience has been accumulated. Closer control over coolant temperatures is now maintained, and the pump is operated routinely at 300 rpm, a speed which experience shows provides sufficient torque to keep the pump running but still does not circulate an excessive amount of sodium through the primary system. Conversely, the 6-in. 1500-gpm-capacity pump in the secondary system has never given difficulty and has operated uninterruptedly for 1000-hr periods that were terminated by shutdown of the system rather than by difficulties with the pump.

The 6-in. 1500-gpm pump in the primary system, on the other hand, was the main offender in the fuel-element damage incident that occurred in July 1959. Difficulties had been experienced from time to time with binding similar to that of the 2-in. pump, and, on two occasions, the auxiliary coolant (Tetralin) in the freeze seal had leaked into the main sodium coolant stream. This was detected by identifying hydrocarbon vapor in the atmosphere above the reactor pool. The first of these occurrences, in April 1958, resulted in an insignificant amount of leakage before the freeze seal was repaired. This leak was caused by a pinhole in the freeze-seal casting. The second such leak occurred in May 1959 and resulted in somewhere between 2 and 10 gal of Tetralin being admitted to the primary sodium stream.[5] This leak was traced to failure of a thermocouple well. The Tetralin-

cooled shaft seal was then replaced with a NaK-cooled seal arranged in such a way that two independent barriers would have to fail before mixing of NaK and the primary sodium could occur. As will be mentioned later, however, sufficient Tetralin had already been admitted to the system to create the condition that damaged the fuel assemblies.

The heat-transfer performance of components outside the core proved to be both enlightening and somewhat disappointing.[6] The main intermediate heat exchanger is a shell and U-tube unit of the simplest configuration deemed feasible in 1954. The straight sections in the shell are baffled, but the expansion space in the U end is unbaffled. Measurements have indicated two primary faults with this design. First, the unbaffled end section is substantially ineffective in heat transfer, and, second, the baffles that direct flow back and forth across the tube bundle horizontally are relatively ineffective in preventing stratification of sodium on the shell side. The high temperature drop taken across this heat exchanger from inlet to outlet, 460°F or greater, causes significant density changes in the sodium, and the cooled sodium sinks to the bottom of the exchanger. This, of course, upsets the temperature distribution, affects the efficiency of heat-transfer surfaces, and also introduces unforeseen thermal-stress patterns. Results of these phenomena are that (1) the heat exchanger is only about two-thirds as efficient as designed, thus requiring a log mean temperature difference across the heat exchanger (temperature drive) of 90°F rather than 60°F, and (2) thermal-stress patterns are set up which are expected to limit useful life in this piece of equipment. In fact, in the vicinity of the tube sheet, "wrinkles" 0.0625 in. in height have already been observed in the shell of the heat exchanger, which is 0.14 in. thick. A new heat exchanger designed to alleviate the problems with the original equipment has been obtained for installation when convenient. Although the new heat exchanger is, in general, similar to the original equipment, it contains additional heat-transfer surface and is baffled to inhibit stratification of cooled sodium on the shell side.

The once-through steam generator procured by the SCEC has, in general, been satisfactory.[7] It, too, is a horizontal U-shaped, shell-and-tube unit, with sodium in the shell and feed water (and steam) in the tubes. Each tube is made up of two concentric tubes, with mercury in the annulus. The mercury serves as a leak detector in case of failure of either of the tubes. Heat-transfer performance has been satisfactory in all but one respect; the stratification noted in the intermediate heat exchanger also exists in the steam generator. The shell side, which contains secondary-system sodium, is essentially unbaffled except for tube supports, and temperature differences of up to 150°F have been measured vertically across the outlet (steam) tube sheet. Apparently saturated steam is being delivered from at least some of the bottom tubes in the area, and highly superheated steam is produced by the top tubes. As these streams mix, a net superheat is achieved, but undesirable stress patterns are set up across the tube sheet. It has been noted that the shell of this unit, which has a developed length of 80 ft, has become bowed to the extent that the outlet end of the heat exchanger has raised 4 in. from its original position, both measurements having been taken with this end unrestrained. On the other hand, no evidence of deposition of foreign materials on the water side has been found, and there have been no indications of chloride stress corrosion, despite construction of the unit from type 304 stainless steel. A rigid program of water purification has been developed and adhered to by SCEC. Total dissolved solids are maintained at less than 0.5 ppm, and oxygen is maintained at an undetectable level through hydrazine injection. Internal inspection of the steam generator during a recent shutdown period indicated that these measures have been highly successful in preventing the development of conditions that caused difficulty in previous sodium-heated steam-generating systems.

Bellows-sealed valves in sizes of 2 in. or less were used throughout the auxiliary systems. These valves have been a source of some difficulty, particularly as a result of bellows failure. Collection of sodium oxide in the bellows area and occasional freezing of sodium in the volume provided by the bellows have interfered with subsequent operation of the valve and have been the prime causes of system failures, as established by post-mortem examination. Nine valves of this type have failed in the primary and secondary systems. None of these failures have occasioned more than inconvenience, since leakage sodium usually freezes, oxidizes, and prevents further leakage from the failed valve; however, the valve is rendered inoperative. These valves are used only in auxiliary systems,

and therefore it has not been necessary to shut down the plant immediately to repair a valve that failed. Such repairs have, however, inevitably extended maintenance shutdown periods.

Conversely, the primary sodium system stop valves, which are sealed by frozen sodium at the stem, have been completely satisfactory. Therefore, as bellows-sealed valves are removed from the auxiliary systems, they are replaced with freeze-sealed valves cooled by ambient inert gas at the gallery temperature (140°F or less). Such valves have been completely adequate in experimental loops containing sodium at temperatures up to 1200°F.

Reactor Structure

The reactor structure is unique in that the required containment is immediately adjacent to the core, and no external sphere or pressure-containment building is provided.[8] This design is based primarily on the following three considerations:

1. The coolant is essentially at atmospheric pressure, and thus it represents negligible stored mechanical energy.
2. The fuel, moderator, and coolant are chemically compatible, and thus there is no potential for release of chemical energy.
3. The system has a large thermal capacity that is available for absorbing any conceivable unplanned release of nuclear energy.

The stainless-steel core tank, which contains the moderator and fuel elements, is surrounded by a second containment vessel, which has dimensions such that, in the event of a leak in the reactor vessel, the sodium coolant will be retained and the level of the sodium in the reactor pool will not drop below the level of the outlet nozzles. An inert-gas atmosphere is maintained in the second containment vessel, and therefore sodium circulation can be continued even in the event of failure of the core tank. Should both these vessels fail, tertiary containment is provided by a carbon steel "cavity liner," which serves to support thermal insulation on its inner surface. The space between the secondary container and the cavity liner is such that a rupture of both the two inner vessels will not drop the sodium pool level below the top of the core. Thus adequate decay heat cooling of the fuel elements would be available in the unlikely event that both inner containers ruptured. Leak detectors are provided to indicate failures of the two inner tanks.

The core tank and the inner containment vessel are cylindrical and are sealed at the top by a rotatable steel plug filled with magnetite ore and concrete grout, with a total weight of 75 tons. This top plug is pierced for accommodation of 81 smaller shield plugs from which fuel elements are suspended, control and safety rods are inserted, and core instrumentation is manipulated. These smaller plugs are sealed with rubber O rings that operate at ambient temperature. The entire shield plug is sealed to the surrounding biological shield by a low-melting metal alloy (Cerrobend) which solidifies in a tongue-and-groove seal. Double containment of the reactor nozzle extends 6 ft from the reactor pressure vessel to a point immediately adjacent to the system stop valves and terminates in a bellows seal that accommodates differential expansion. An inert-gas atmosphere is maintained within this containment space, and leak detectors are provided to indicate a pipe or nozzle failure. Sodium is conveyed in single-walled piping outside these bellows, with secondary containment provided by concrete shielding blocks over the galleries. The shielding blocks are sealed with neoprene gaskets, and an oxygen-deficient atmosphere (less than 1 per cent O_2) is maintained inside the galleries during reactor operation.

No failures have been detected in piping or containers by leak detectors or other indications, such as loss of pressure or sodium smoke. The neutral atmosphere maintained in the galleries proved effective in preventing sodium combustion in the one instance of a leak in a 2-in. bellows valve in the secondary system. The leak was first indicated by a wisp of sodium oxide "smoke" from reaction with air in the valve operator. Only static sodium was involved, and there was no radioactivity associated with the smoke. In this case the sodium was frozen in the line by turning off resistance heaters, and operation proceeded until the next scheduled shutdown period, at which time the leaking valve was cut out and replaced. Examination of the valve indicated that less than 1 lb of sodium had leaked out and that this sodium, after solidification and oxidation, had effectively sealed the leak. The valve was, of course, rendered inoperative. Other valves in the secondary (nonradioactive) system had leaked from time to time, with generally similar results

(i.e., sodium fires did not occur because of solidification and the formation of a protective crust over the free sodium).

Difficulty was experienced early in the reactor outlet nozzle area.[7] This nozzle conducts sodium at the outlet temperature of 960°F to the main circulating piping system, and it was not provided with a thermal baffle. The tank wall to which the nozzle was attached was baffled to maintain approximately 1 in. of stagnant sodium against the tank wall in order to alleviate temperature gradients. When power operation was initiated, it was found that the circulation of sodium by natural convection after a scram was approximately 30 per cent greater than originally calculated. If the circulation were not controlled, the fuel elements would be cooled more rapidly than decay heat was being generated, and the temperature at the top of the sodium pool would be rapidly lowered. This created thermal stresses in the nozzle, since the nozzle assumed the new sodium temperature; whereas the wall, which was thermally baffled, retained its original temperature for a significant period of time. This did not create a problem during startup, steady-state operation, or normal shutdown operation, but calculations showed that it would cause excessive stresses during a scram from full power. Consequently, the reactor power level and the temperature were limited to values that could not induce stresses past the yield point of type 304 stainless steel until an eddy-current brake could be installed in the primary system to control convective sodium flow after a scram. Temperature measurements taken after installation of the eddy-current brake have indicated that the brake is completely successful in matching flow and heat-generation characteristics following a scram at full power.

Radiation Levels

The primary cooling system of the SRE is, of course, contained in biological shielding of concrete aggregate to control radiation from the Na^{24} generated in passage of the coolant through the reactor. The buildup of radiation on system piping through the various agencies was, however, of great interest in terms of accessibility after periods of reactor operation. It was considered that the temperature conditions of the primary sodium system (i.e., a maximum temperature of ~1000°F and a temperature difference across the system of 500°F) were potentially conducive to accelerated mass transfer of iron, cobalt, nickel, and other constituents of the stainless-steel and zirconium components of the circulating system. In addition, the possibility of fission-product leakage is always present.

No activity transport could be measured during the first two years of reactor operation, and during that time sodium was circulated in the system for approximately 16,000 hr, 4500 hr of which was at a high temperature and a large temperature differential. During this two-year period, two fuel elements were found to have pinhole leaks, and there was some release of gaseous fission products. These were detected by the presence of Xe^{133} in the reactor helium atmosphere, as indicated by a pulse-height analyzer. The leaking fuel elements were identified by withdrawing the elements into the fuel-handling cask during a shutdown period and noting any increase in gas activity within the cask. The two fuel elements which were defective revealed their presence quite clearly by immediately producing an increase in the count rate measured by the cask-gas sampling apparatus from a background value of 200 dis/min to count rates of 3000 and 16,000 dis/min, respectively. The remaining 41 elements caused no increase in count rate upon sampling of the cask gas. Visual examination of the suspect fuel elements in the SRE hot cell indicated no defects visible to binocular-aided vision, but these elements were removed from the reactor and stored for eventual reprocessing.

A massive release of fission products throughout the primary system occurred in the July 1959 incident, as described previously.[2,3] Thirteen of the 43 fuel assemblies sustained damage sufficient to cause failure of the cladding on all seven rods of the assembly and to form at least some iron-uranium eutectic that was molten for a short period of time in the reactor. Measurements of cover-gas activity extrapolated to the time of the fuel-element damage indicate that approximately 10,000 curies of mixed fission-product activity was released to the coolant stream, although this massive release of activity remained undetected for some days because of the complete containment afforded by the SRE shielding system. This occurrence, which caused some fuel-element temperatures to rise abnormally high, at least to 1430°F, came about during a period

of reactor operation in the 2- to 3-Mw(t) range. This power level was being maintained in order to alleviate a temperature maldistribution throughout the reactor, which was considered, on the basis of previous similar experience, to be caused by impurity collection on the orifice plates. Temperature-induced reactivity fluctuations amounting to 0.3 per cent $\Delta k/k$ were found upon examination of records after the fuel-element damage was discovered. The operators noted subjectively that the reactor was somewhat "touchy" to control, and, when the reactor went on a 7-sec positive period, it was immediately shut down. A cautious restart did not reproduce any of the symptoms present prior to the fast period, and low-power operation was continued for several days. These events were described previously[2] and have been analyzed by Fillmore.[9]

The primary piping system was opened three months after the release of radioactivity to the system. At that time, the radiation level throughout the primary gallery averaged 500 mr/hr, but isolated spots in the system gave readings as high as 15 r/hr at the piping surface. These spots were almost invariably associated with irregularities in the system, such as thermocouple wells, heat-exchanger end bells, and horizontal piping runs containing sodium that was normally circulated at a low velocity. Cold trapping through a 30-gpm side-stream loop was undertaken, and it was apparent that fission products were being removed by the cold trap, since the radiation level rose from substantially zero to 190 r/hr at a specific point on the surface of the trap in a period of six days. At this time, the cold trap was removed because the radiation level was increasing beyond that which could be handled by the available shielding. A new cold trap was installed and monitored, and a similar increase in activity from zero to about 90 r/hr occurred in five days. While the cold trapping was being performed, sodium was being circulated at the maximum flow rate of 1500 gpm through the primary system. After the cold-trapping period, the radiation levels in the pipe gallery had dropped to an average of 150 mr/hr. By applying lead-sheet shielding locally in the region of the cold leg of the intermediate heat exchanger and at some piping depressions, the radiation level was reduced in working areas to the 8-hr tolerance.

Upon draining the sodium from the reactor, it was discovered that a black sponge-like material, which had apparently been floating on top of the sodium pool, had been deposited on the tops of the moderator cans. Later analysis proved this substance to be amorphous carbon with about 25 per cent sodium and sodium oxide in the vacant holes. Visually, this mass had the appearance of black foam with small spherical bubbles that ranged from 0.1 to 1.0 mm in diameter. Radiation analysis of this deposit, which was removed with a vacuum device operated through hermetically sealed shield plugs in the loading face shield, indicated that much of the activity was concentrated in the carbon. The activity of sodium samples taken six months after the fuel-element damage was 0.3 $\mu c/cm^3$, but the carbon activity was 0.03 curie/cm^3, a factor of 10^5 greater. The specific carbon activity was difficult to determine because of the spongy nature of the material, but the orders of magnitude are considered to be correct. Vacuum cleaning of the tops of 100 of the 119 moderator cans removed approximately 10 lb of carbonaceous, sodium-bearing material in which the bulk of the fission-product activity was apparently concentrated. The remaining carbonaceous material is currently being removed by filtration in the cold trap and by hot trapping the dissolved carbon.

The specific activity of the sodium is declining in accordance with the decay curve for mixed fission products. The sodium has been reintroduced into the reactor, but the maximum temperature has been limited to 800°F because of the dissolved carbon. Concern is felt that, at temperatures greater than 800°F, the dissolved carbon might cause carburization of the 0.010-in.-thick type 304 stainless-steel fuel cladding.[10] Higher temperatures will not be generated until the cold and hot traps have reduced the dissolved carbon content of the sodium to 20 ppm or less.

Analysis of the helium cover gas shortly after discovery of the activity release showed that only the noble gases were released. Xenon and krypton were identified as the sole contributors to cover-gas activity. No I^{131} has been detected except that combined with sodium. Removal of the irradiated fuel from the reactor reduced the cover-gas radioactivity to near normal background, despite the presence of fission products in the sodium. Disposal of piping and components replaced during the recovery period has not proved to be a severe problem. The normal methods used for cleaning components containing uncontaminated sodium were applied, with

the additional precaution of retaining the sodium-steam reaction products which apparently contained all the fission products not deposited on pipe walls. Fission products remaining on pipe walls created radiation levels of only a few milliroentgens per hour, and these components could be disposed of as relatively low-level wastes by conventional methods. The reaction products were disposed of as low-level liquid waste by diluting and mixing with concrete.

Repair Under Radioactive Conditions

For the first nine months of the repair period, exposed, irradiated fuel elements were present within the reactor structure. This necessitated performing all operations on the core structure through hermetic seals, since the core cover-gas activity ranged from 0.3×10^{-2} to 1.0×10^{-2} $\mu c/cm^3$. Inasmuch as the activity was well above the atmospheric release tolerance, complete containment was required. Inflatable seals were found to be adequate and were widely employed in the equipment designed and constructed for removal of damaged fuel elements, moderator cans, fuel slugs, and other debris that had been deposited in the core. In addition, it was necessary to prevent the spread of activity over the reactor floor by repair equipment which had been mated to the reactor face and through which irradiated material had passed. Vinyl plastic sheet, laid over the entire floor to prevent contamination of the concrete, performed this function effectively. Regular replacement was required because of wear and tear, and the sheet was necessarily disposed of as contaminated material. Cloth and plastic booties, used in conjunction with coveralls, were employed by the operating crews involved in reactor floor operations. Continuous monitoring of personnel did not reveal any instance of contamination. Work in the primary-system pipe galleries, which involved cutting and welding, was performed by Atomics International craftsmen. Components that required replacement were cut from the system by men using respirators; the area adjacent to the cut was thoroughly cleaned of sodium by using butyl alcohol prior to conducting rewelding operations. These operations were carried out under the supervision of health physicists; again, no activity release above tolerance was noted, and no person received more than 300 mr/week, despite the presence of contaminated equipment and a rather high background activity level from piping and system components.

Operations within the reactor core were particularly time consuming because tools and viewing apparatus had to be improvised for each situation. The operations were further complicated by the necessity for keeping the entire reactor structure at a temperature above 250°F in order to prevent solidification of any residual sodium that might be adhering to moderator cans or might be trapped by carbonaceous deposits. Mercury-vapor lamps, built into shield plugs 3 in. in diameter, were inserted into the top shield for illumination, and various remotely controlled grappling tools were employed to remove 82 fuel slugs 6 in. long and ¾ in. in diameter from the top of the core where they had fallen upon removal of the partially severed elements. A particularly difficult situation was presented when inspection of the bottom plenum (33 ft from the operating face of the reactor), with the use of locally built periscopes and illumination, revealed two fuel slugs. It was necessary to devise a tool capable of penetrating 33 ft from the reactor face through a minimum opening of 2 in. and then reaching 18 in. to the position of the slugs. A "clamshell" device was designed to recover these slugs so that, in the event they were dropped, there would be no possibility of their shattering in the bottom plenum. Shattering was considered a good possibility if the slugs were dropped because uranium-iron eutectic is quite brittle, and there was no assurance that it was not present to some degree on these two slugs. Indeed, slugs removed from the top of the core were found upon hot-cell examination to be extremely brittle. The entire recovery operation was quite tedious, but a thorough inspection of all available parts of the reactor structure indicated that all fuel material was removed.

System Kinetics

The period of operation prior to fuel damage demonstrated the stability of the reactor, which has a prompt negative power coefficient of 2.2×10^{-5} $(\Delta k/k)$/Mw(t), and a positive moderator temperature coefficient of approximately 1×10^{-5} $(\Delta k/k)$, with a time constant of 10 min at normal operating temperatures.[11] The over-all power coefficient at 20 Mw(t) is -7×10^{-5} $\Delta k/k$ per thermal megawatt. The reactor is stable under all normal operating conditions, and it

follows power swings of 4 Mw(t)/min (20 per cent per minute) induced by the SCEC steam plant. An automatic control system has been installed so that reactor operation can be completely controlled by the turbine operator; this system will be programmed from a remote station when electric power generation is resumed in 1961.[12] Careful measurements have indicated no power shift resulting from xenon oscillations, and application of existing theory demonstrates that such power shifts should not be expected in sodium-graphite reactors of any conceivable size. The fuel irradiation lifetime of the first core loading averaged only 1100 Mwd/ton, and therefore changes in the fuel temperature coefficient with irradiation were not measurable. Reactivity losses from burnup and fission-product production were consistent with calculations,[13] although they were of minor absolute magnitude, since burnup proceeded to only a maximum of 0.15 at.%.

The radial power distribution across the SRE core would cause significant temperature differences between outlet fuel channels if orificing were not employed. Orificing is, in fact, employed to maintain the outlet temperatures from each channel within 50°F of the desired mixed-mean outlet temperature. The orifice plates are readily changeable in a hot cell. Prediction of orifice sizes has become quite accurate with experience, and, as the power profile changes with control-rod position and total power generation, orificing may now be predicted for full-power operation with confidence.

Pile oscillator techniques have been extensively used in determining stability.[14] Separation of fuel, coolant, and moderator power and temperature coefficients has been demonstrated, as well as over-all stability of the reactor under any conceivable operating condition.

Even under extremely adverse conditions, such as those present immediately prior to the fuel damage, the reactor was completely controllable. Although the moderator temperature apparently rose significantly, no runaway tendencies were observed. Completely stable operation of the reactor was observed at all times, except during the few minutes in which the actual fuel damage occurred. This damage apparently resulted in clearing blocked channels to some extent and permitted stable temperatures to be maintained in the core structure despite the existence of severely damaged cladding. A complete discussion of the reactivity fluctuations has been prepared.[9]

Conclusions

Operation of the SRE for three and one-half years, including a period of recovery from extensive fuel damage, has demonstrated the following:

1. The reactor system is both stable and responsive to power changes.
2. Maintenance and repair of sodium systems uncontaminated by fission products is a routine procedure.
3. Release of fission products to the primary system of a sodium-graphite reactor does not result in hazards to operating personnel or to the general public when the SRE containment philosophy is employed; the sodium apparently traps the bulk of fission products released from the fuel.
4. Detection of fission-product release by monitoring the cover gas is effective.
5. Repair of contaminated sodium reactor systems is feasible.
6. Prediction of sodium-graphite reactor behavior, both in normal and abnormal situations, has been confirmed.
7. Special containment measures, other than those currently employed, are unnecessary.
8. The sodium-graphite reactor concept has been demonstrated, by operation of the SRE, to be technically feasible.
9. The SRE is conservatively designed with respect to power capability.

Organization and Procedures in Licensing and Regulation

The AEC Division of Licensing and Regulation has been reorganized and expanded, as shown in Fig. VI-1, in a move that was necessitated by the growth of the Commission's regulatory program. The new organization of the Division of Licensing and Regulation reflects AEC emphasis on the development of safety standards in the peaceful use of atomic energy. The Division is principally responsible for carrying out AEC staff functions concerning Commission licenses.

The former Hazards Evaluation Branch (HEB) has been divided into five branches which comprise the subdivision Nuclear Facilities Safety, which is headed by C. K. Beck, who was for-

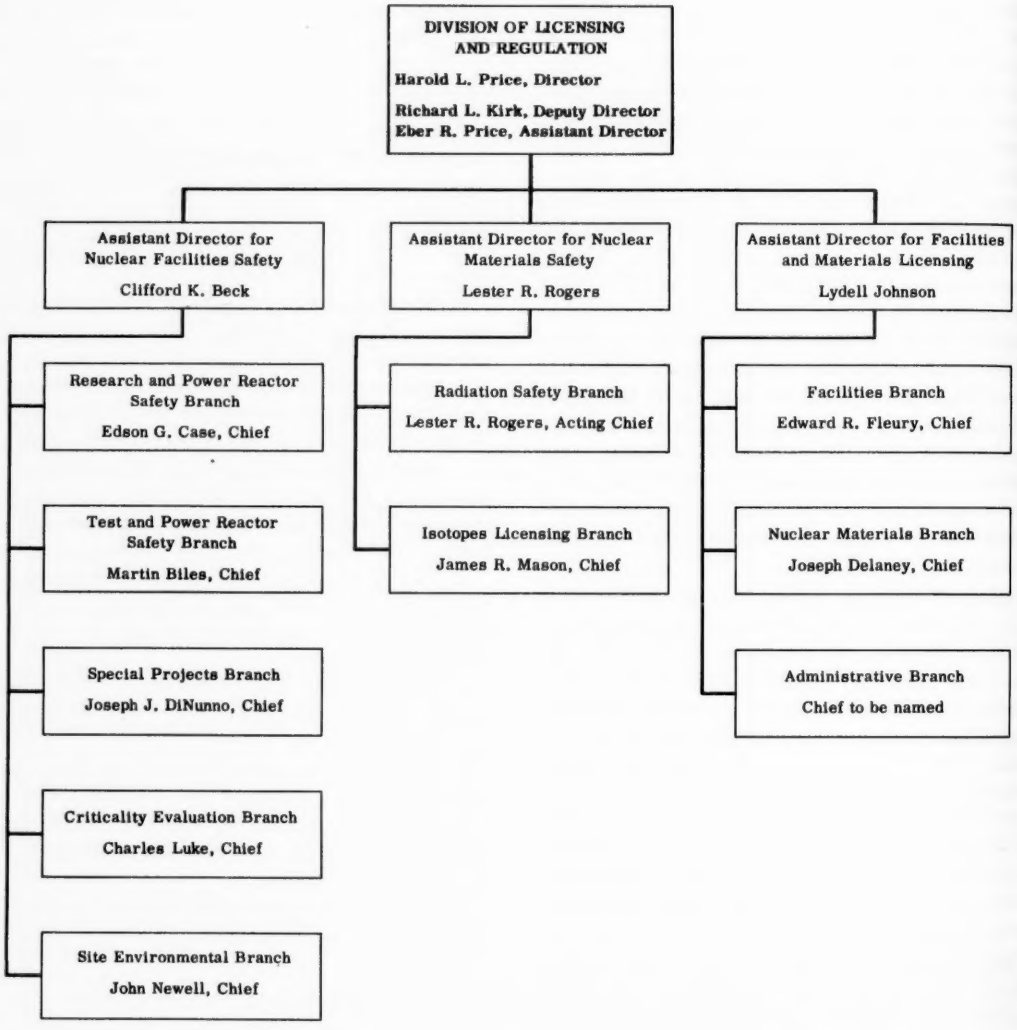

Fig. VI-1 Organization chart of the AEC Division of Licensing and Regulation.

merly Chief of the HEB. The functions of each of the five new branches are described in the following statements:

1. The Research and Power Reactor Safety Branch is responsible for the hazards evaluation of boiling-water, pressurized-water, graphite-production, organic-moderated, research, and training reactors, as well as critical assemblies.

2. The Test and Power Reactor Safety Branch is responsible for the hazards evaluation of sodium-cooled, gas-cooled, heavy-water-production, and test reactors, as well as for administering examinations for reactor operator licenses.

3. The Special Projects Branch is responsible for developing safety standards, criteria, and guides for the location, design, construction, and operation of reactors. Other responsibilities of this branch relate to the transportation of nuclear material, hazards summary report standardization, aerospace activities, evaluation of nuclear incidents, research programs in nuclear safety, and preparation of technical data.

4. The Criticality Evaluation Branch is responsible for the technical review of facilities

and material licenses and for the review of the provisions for the transportation of nuclear materials to assure against accidental criticality.

5. The Site Environmental Branch is responsible for the safety evaluation of proposed nuclear facilities and sites, with respect to meteorology, hydrology, seismology, and waste disposal, and is further responsible for liaison with federal and state agencies in these areas.

Under L. R. Rogers, Assistant Director for Nuclear Materials Safety, will be the existing Radiation Safety Branch and the Isotopes Licensing Branch. The functions of these branches are not changed. The Radiation Safety Branch develops and recommends radiation-protection control policies, procedures, and criteria, and the Isotopes Licensing Branch technically evaluates and licenses the use of by-product materials.

The former Licensing Branch, under L. Johnson, has been divided into two branches which he will direct. They are (1) the Facilities Branch, which will be responsible for reactor licensing, for export of reactors, and for issuing operators licenses, and (2) the Nuclear Materials Branch, which will have charge of licensing applicants to receive, possess, use, and transfer natural uranium and thorium, U^{235}, U^{233}, and plutonium. The new Administrative Branch will be responsible for administering the program for indemnification of licensees against public liability arising from nuclear incidents.

In addition to reorganizing the Division of Licensing and Regulation, the AEC has issued two amendments to its licensing regulations to facilitate the issuance of operating licenses for civilian power reactors and to clarify the extent to which construction of a reactor facility may proceed prior to issuance of a construction permit. The amendments became effective in October 1960, as part of Part 50, Title 10, *Code of Federal Regulations*.

The first amendment (Section 50.10*b*) is designed to clarify the extent to which construction of a new facility may proceed before a construction permit is issued. The amendment expressly prohibits the laying of the foundation of any portion of the permanent facility at the site. The amendment does, however, permit site exploration, excavation, and preparation, including construction of roadways, railroad spurs, and transmission lines; the procurement or manufacture of components of the facility; construction of nonnuclear facilities such as turbine-generators, turbine buildings, and temporary buildings for use in connection with the construction of the facility; and the construction of a building, such as a college laboratory, which is to house a medical or research reactor, but which is also to be used for other purposes.

The second amendment, which adds a new Section 50.57, establishes criteria and procedures for the issuance of a provisional operating license for a reactor where construction cannot be completed before fuel loading or where review of operating experience is deemed advisable before a full-term license is granted to an applicant. Such a license will be issued if the AEC finds (1) that there is reasonable assurance that the facility will be completed in conformity with the construction permit and application, as amended, (2) that the activities authorized by the provisional operating license can be conducted without endangering public health and safety, and (3) that the facility will be ready for initial fuel loading within 90 days from the date of issuance of the provisional license. The duration of provisional licensing amendments may not exceed 18 months except where, upon good cause, the expiration date may be extended. (W. B. Cottrell)

Action on Reactor Projects by Licensing and Regulating Bodies

In the course of securing final approval for the operation of a power reactor, the applicant must clear several hurdles designed to assure the AEC, and thus the public, that the safety of the public has been adequately considered in a responsible manner. The procedure involves two major steps, as described in Title 10, *Code of Federal Regulations:* (1) securing a construction permit and (2) securing an operating permit.

The processes involved in obtaining a construction permit and in obtaining an operating license require the preparation of a hazards summary report that must be reviewed and approved by both the Division of Licensing and Regulation and the Advisory Committee on Reactor safeguards (ACRS). Before each license is issued, a public hearing is held. As a consequence of these various administrative reviews, any given reactor may be involved in reviews for a period of years, since the safety review process must continue through all the stages of

Table VI-1 CALENDAR OF LEGAL STEPS IN LICENSING U. S. POWER REACTORS

Reactor information							Construction permit						
			Max. power		Hazards report								
Code No.	Name, operator, and location	Reactor type	Mw(t)	Mw(e)	No.	Date	Findings of ACRS-HEB or AEC staff	Findings of hearing examiner	Permit No.	Permit Date	Scheduled completion date	Additional information	Operating license (and other information)
PP-4	BONUS, Puerto Rico Water Resources Authority, Punta Higüero, Puerto Rico	Boiling H_2O nuclear superheater	50	173	PRWRA-GNEC-2	12-21-59	3-14-60, ACRS, favorable; 6-60, AEC, favorable	6-28-60, favorable	Not required		9-62	Construction authorized effective 7-19-60	
50-144	CVTR, Carolinas-Virginia Nuclear Power Associates, Parr, S. C.	Pressurized D_2O moderator and coolant	605	17	CVNA-27 (3 vols.)	7-15-59	12-11-59, AEC, favorable	4-12-60, favorable	CPPR-7	5-4-60	1-1-62 to 12-31-62		
50-155	Big Rock Point, Consumers Power Co., Charlevoix County, Mich.	Boiling H_2O	240	75	Not numbered	1-60	4-18-60, ACRS, favorable	4-18-60, favorable	CPPR-9	5-31-60	7-1-62 to 7-1-63		
50-10	Dresden, Commonwealth Edison Co., Grundy County, Ill.	Boiling H_2O	630	192 gross, 180 net	Not numbered (4 vols.) Summary	Through 9-57 1-6-59			CPPR-2	5-4-56		Construction began 3-57, completed 9-59	License No. DPR-2 for half power is dated 9-26-59 and expired 12-10-59; license for temporary full power is dated 6-9-60; license for 40-yr term full power is dated 10-14-60 and expires 5-4-96
PP-1	Elk River, Rural Cooperative Power Associates, Elk River, Minn.	Boiling H_2O	58	22	Not numbered (rev.) Suppl. 1	3-12-59 5-25-59		1-19-60, favorable; 4-60, favorable	Not required				Hearing is scheduled for 3-7-61; ACRS findings were favorable 11-5-60
F-16	Enrico Fermi, Power Reac-	Fast-neutron Pu breeder,	300	104 gross, 94 net	APDA-108 APDA-117	9-1-55 10-15-56			CPPR-4	8-4-56	7-15-61	Validity of con-	
									Reaffirmed 12-10-58				

	Power Associates, Elk River, Minn.				(rev.) Suppl. 1	5-25-59		4-60, favorable				findings were favorable 11-5-60
F-16	*Enrico Fermi, Power Reactor Development Co., Lagoona Beach, Mich.*	*Fast-neutron Pu breeder, Na cooled*	*300*	*104 gross, 94 net*	APDA-108 APDA-117 APDA-115 APDA-124 Exhibits through No. T18	9-1-55 10-15-56 11-1-56 1-59 Through 9-10-59			CPPR-4	8-4-56 Reaffirmed 12-10-58 Reaffirmed 5-26-59	7-15-61	Validity of construction permit is before U. S. Supreme Court
	EGCR, Tenn. Valley Authority, Oak Ridge, Tenn.	He cooled, graphite moderated; fuel testing	84	25	ORO-196	5-19-59	5-59, ACRS, favorable		Not required		12-31-62	Construction approved and in progress
50-152	FWCNG, Florida West Coast Nuclear Group, Polk County, Fla.	D_2O moderated, CO_2 cooled	153	50	GNEC-115 (rev.) (3 vols.)	12-7-59	Application pending					Restudying costs
PP-3	HNPF, Consumers Public Power District, Hallam, Nebr.	Graphite moderated, Na cooled	240	75 gross	NAA-SR-3379 CPPD-25 NAA-SR-4504	No date 9-22-55 No date	7-25-59, ACRS, favorable; 2-8-60, ACRS, favorable; 4-29-60, AEC, favorable	6-6-60, favorable	Not required		7-62	Construction authorized effective 6-28-60
50-133	Humboldt Bay, Pacific Gas & Electric Co., Humboldt Bay, Calif.	Boiling H_2O, vapor suppression (safety)	163	60 gross	Not numbered	4-15-59	7-60, ACRS, favorable	10-17-60, favorable	CPPR-10	11-9-60	1-1-62 to 7-1-63	
50-3	Indian Point, Consolidated Edison Co. of New York, Buchanan, N. Y.	Pressurized H_2O; Th converter	585	275 gross 255 net	Exhibits through L-4 Exhibits through K-4 Exhibit K-5 (Rev.1)	Through 1955 Through 1958 1-60			CPPR-1	5-4-56	10-1-60 to 10-1-61	

(Table continues on next page.)

Table VI-1 (Continued)

Reactor information							Construction permit						
			Max. power		Hazards report								
Code No.	Name, operator, and location	Reactor type	Mw(t)	Mw(e)	No.	Date	Findings of ACRS-HEB or AEC staff	Findings of hearing examiner	Permit No.	Permit Date	Scheduled completion date	Additional information	Operating license (and other information)
	N.S. *Savannah* Maritime Administration, U. S. Dept. of Commerce	Pressurized H_2O	69	22,000 shp	BAW-1164	7-60			Not required			Unique legal status; Presdential recommendation; Congressional authorization	Critical tests and nuclear-powered trial runs scheduled in 1961
50-130	Pathfinder, Northern States Power Co., Sioux Falls, S. Dak.	Boiling H_2O nuclear superheater	203	66 gross, 62 net	ACNP-5905 ACNP-5917	3-10-59 8-31-59	3-60, ACRS, favorable	4-60, favorable	CPPR-8	5-12-60	1-1-62 to 7-1-62		
50-171	Peach Bottom, Philadelphia Electric Co., Conowingo Pond, York County, Pa.	High temperature; unclad fuel; gas cooled; graphite moderated	115	40	Not numbered	12-59	3-14-60, ACRS, favorable; 12-10-60, ACRS, site favorable						
PP-2	Piqua OMR, City of Piqua Municipal Power Commission, Piqua, Ohio	Organic moderated, organic cooled	46	114	NAA-SR-3100SP NAA-SR-3279 NAA-SR-3575 NAA-SR-4048	9-28-58 No date 4-13-59 7-1-59	7-25-59, ACRS, favorable	1-7-60, favorable				Continuance of construction authorized effective 1-28-60	
	PWR, Duquesne	Pressurized	231	68 gross,	WAPD-SC-	6-57						Construction	Operating at full

	Piqua, Ohio			3575 NAA-SR-4048	7-1-59							five 1-28-60
	PWR, Duquesne Light Co., Shippingport, Pa.	Pressurized H_2O	231	68 gross, 60 net	WAPD-SC-547 WAPD-SC-541 Other supplements	6-57 9-57 11-57					Construction started in May 1955	Operating at full power since 12-23-57; new seed element authorized; critical 4-12-60, on line 5-6-60; ACRS findings were favorable 11-4-57
50-146	Saxton, Saxton Nuclear Experimental Corp., Berks County, Pa.	Pressurized H_2O	20	5 net	Not numbered	No date	1-7-60, ACRS, favorable	1-21-60, favorable	CPPR-6	2-11-60	12-1-61 to 12-1-62	License is to expire 2-11-66 or five years after criticality, whichever is earlier
50-18	VBWR, General Electric Co., Pleasanton, Alameda County, Calif.	Boiling H_2O test	30* 50*	3	GEAP-069 SG-VAL-2 Amendments	10-17-55 8-31-57 11-12-57, 1-30-59			CPPR-3	5-14-56	2-28-58	License No. DPR-1 is dated 8-31-57 and expires 5-14-63; operating since 9-57; operating at 30 Mw(t), 50 Mw(t) authorized
50-29	YAEC, Yankee Atomic Electric Co., Rowe, Mass.	Pressurized H_2O	392	145 gross, 136 net	Not numbered	No date		CPPR-5	11-4-57			ACRS findings on 1-30-60 and 5-9-60 were favorable; findings of the hearing examiner were favorable on 5-17-60; license No. DPR-3 is dated 7-9-60 and 7-29-60 provisional 5-Mw(t) and full-power tests; critical 8-19-60

NOTE: BONUS = Boiling Nuclear Superheater; CVTR = Carolinas-Virginia Tube Reactor; EGCR = Experimental Gas-Cooled Reactor; FWCNG = Florida West Coast Nuclear Group; HNPF = Hallam Nuclear Power Facility; PWR = Pressurized-Water Reactor; VBWR = Vallecitos Boiling-Water Reactor; YAEC = Yankee Atomic Electric Co.

*Operating at 30 Mw(t); 50 Mw(t) is authorized.

plant design and construction up to the time of routine on-line operation. These reviews frequently result in modification of the reactor design or operation to enhance safety, and, in any case, they constitute the precedents for "adequate" safety.

Most power reactors proposed for construction and operation in this country are currently involved in some stage of the review process, as indicated in Table VI-1. Furthermore, in order to schedule its own review activities to be compatible with the construction schedules and progress of the applicant, the Division of Licensing and Regulation has undertaken to establish a working schedule for the review of major reactor projects. The tentative schedule for these projects is presented in Table VI-2. These tentative schedules for either the construction license or operating license are based, respectively, on the applicant's date for start of construction or for initial criticality. These dates are flexible and may be expected to change to reflect either significant changes in the progress of the work on the facility or delays in the review process.

Vallecitos Boiling-Water Reactor

(Docket 50-18)

The General Electric Company (GE) Vallecitos Boiling-Water Reactor[15] is a 20-Mw boiling-water reactor that was first operated in 1957. The reactor, which is now in operation, was shut down in October 1959 for modifications

Table VI-2 WORKING SCHEDULE FOR REVIEW OF LICENSE APPLICATIONS

(For a More Detailed Identification of Each Reactor Facility, See Table VI-1)

	CVTR	Big Rock Point	Path-finder	Hum-boldt Bay	Saxton	Piqua	Hallam	Peach Bottom	BONUS	Elk River	Indian Point
Date schedule established	8-60	8-60	8-60	8-60	8-60	8-60	8-60	9-60	9-60	9-60	11-60
Preliminary hazards report submitted											
Reviewed by Division of Licensing and Regulation								10-60			
Preliminary hazards report reviewed by ACRS								11-60			
Public hearing								11-60			
Intermediate decisions of hearing				9-60				1-61			
Construction permit issued				10-60				2-61			
Start of construction				10-60				4-61	8-60	11-59*	
Completion of preliminary design	4-61		11-60				12-60				
Final safeguards report submitted	10-61	12-62	8-61	10-61	2-61		4-61	4-63	2-62	8-61	9-60
Reviewed by Division of Licensing and Regulation	12-61	2-63	10-61	12-61	4-61	12-60	6-61	6-63	4-62	10-61	12-60
Reviewed by ACRS	1-62	3-63	11-61	1-62	5-61	1-61	8-61	7-63	5-62	11-61	5-61†
Prehearing inspection	2-62	4-63	12-61	1-62	6-61	2-61	8-61	8-63	5-62	12-61	5-61
Hearing on provisional operating license	2-62	4-63	12-61	1-62	6-61	2-61	8-61	8-63	5-62	12-61	5-61
Intermediate decisions of hearing	4-62	6-63	2-62	3-62	8-61	3-61	9-61	10-63	8-62	2-62	7-61
Provisional operating license	4-62	6-63	2-62	3-62	8-61	3-61	9-61	10-63	8-62	2-62	7-61
Completion of construction	6-62	8-63	1-62	3-62	8-61	4-61	10-61	10-63	9-62	1-62	7-61
Operator's license examinations	6-62	8-63	1-62	3-62	8-61	4-61	10-61	10-63	9-62	2-62	7-61
Fuel loading and initial criticality	7-62	9-63	3-62	6-62	11-61	5-61	12-61	12-63	10-62	3-62	9-61

*A five-month delay in the review for the operating license is provided to permit a review, a hearing, and a decision on approval of the technical specifications and their incorporation into the construction permit.

†Subsequent to the approval of this construction permit, a review and a hearing on amendments to the permit were scheduled from August 1960 to February 1961.

which were undertaken to permit the reactor to be used for advanced experimental test programs relating to other nuclear plants with which GE is concerned. As noted in the December 1960 issue of *Nuclear Safety*,[16] the operating license was amended, following a public hearing, so that all changes not covered in the license required prior written authorization by the Director of the Division of Licensing and Regulation. GE filed exceptions[17] to this procedure and also to a requirement that an immediate written report be made of any event of a possibly unsafe nature relating to the operation of the facility. The AEC staff supported GE in the first instance but not in the latter. Both groups maintained that GE could make certain changes to the facility without decreasing the safety, and GE further contended that the procedure created serious delays both for the licensee and the Division of Licensing and Regulation. The argument with respect to written authorization for changes was heard before the AEC hearing examiner, who, in November 1960, ordered[18] that GE's license be amended, as proposed by the AEC staff, to provide GE "complete freedom to make changes within the parameters of the technical specifications, provided no unresolved safety question is involved."

Humboldt Bay Power Plant (Docket 50-133)

The Humboldt Bay Power Plant,[19] which the Pacific Gas and Electric Company has been licensed to construct on the California coast, is a 163-Mw(t) boiling-water reactor and is the first stationary power reactor to employ pressure-suppression containment. The construction permit was issued in November 1960 after approval by the ACRS, as described in the December 1960 issue of *Nuclear Safety*,[20] and the subsequent public hearing.[21] At this uncontested hearing, the chief mechanical engineer discussed the company's atomic development program, financing of the plant, and the unique containment. The AEC testimony by M. Booth consisted of a discussion of several features of the plant relative to safety, including the Zircaloy-2 cladding of the fuel elements, the control-rod drive mechanisms, the pressure-suppression containment scheme, and containment isolation. The Zircaloy-2 cladding will be subjected to continuing tests because of concern for the propagation of small defects. There is concern because, when a rod becomes separated from its drive unit and falls downward under the influence of gravity, it adds reactivity to the core. Further tests are to be carried out. The AEC witness reported favorably on the containment system tests and upon the technique of employing two valves in series to seal all lines penetrating the containment.

Also filed in the record of this facility is a report (entitled "An Evaluation of Possible Effects on Fish and Shellfish of the Operation of a Nuclear Reactor at the Humboldt Bay Power Plant of the Pacific Gas and Electric Company Located near Eureka, California," by W. A. Chapman) in which it is concluded that: "In view of our present knowledge, the discharge of radioactive wastes into Humboldt Bay in the volumes and levels of radioactivity estimated in the request would not result in concentrations of radioactivity that would make marketing of oysters and clams hazardous."[22] However, the basis for this conclusion was subsequently questioned by J. W. Hedgpeth, of the Pacific Marine Station, who noted that no hydrographic work had been done at Humboldt Bay and that the conclusion lacked a quantitative basis.[23]

Although many questions must yet be resolved before an operating permit can be issued, issuance of the construction permit was approved by the hearing examiner in an intermediate decision released in October 1960. The construction permit was subsequently issued in November 1960.

Plum Brook Reactor Facility (Docket 50-30)

The Plum Brook Reactor Facility contains a 60-Mw(t) research reactor that is light-water cooled and moderated and has a primary beryllium reflector and a secondary water reflector.[24] This facility is to be used by the National Aeronautics and Space Administration (NASA) in the development of a nuclear reactor system for aeronautical and space applications. Following a favorable review by the ACRS on the proposed operation of this facility, as discussed in the September 1960 issue of *Nuclear Safety*,[25] a public hearing on the issuance of an operating license was scheduled for October 1960. NASA subsequently requested a postponement of this hearing date because additional time was needed to complete construction.[26] NASA also disclosed that it intended to operate the reactor at 100 kw before completion of all facilities for operation at 60 Mw. This proposal was subsequently re-

viewed[27] by the ACRS, which, at its November 1960 meeting, suggested "that the overpower scram be reset at just above the power limit requested and that one or more temporary instruments be inserted into the core so that water temperature can be observed during the time when the normal instrumentation is not useful and normal coolant flow is not available." The ACRS further concluded "that there is reasonable assurance that the proposed operations [at 100 kw] can be carried out without undue hazard to the health and safety of the public."

Yankee Nuclear Power Station

(Docket 50-29)

The Yankee Nuclear Power Station contains a pressurized-water reactor designed to operate at 485 Mw(t). As discussed in the December 1960 issue of *Nuclear Safety*,[28] a license authorizing operation at power levels not to exceed 392 Mw(t) was issued, and, in November 1960, the Yankee Atomic Electric Company (YAEC) announced that it was producing power at about this level. Previously the Company had filed exceptions[29] to the hearing examiner's decision granting the license in that it included the stipulation: "No ... amendment to the license application shall be effective to change or modify the technical specifications until written approval thereof by the Commission [is] issued after a public hearing on each amendment to [the] Yankee application." Although this exception was denied at the time, in November 1960, when the AEC approved a change procedure for the Vallecitos Boiling-Water Reactor (see article above), the YAEC submitted a motion to reconvene[30] for the purpose of similarly amending the Yankee license so far as applicable and appropriate. Action has not yet been taken on this motion.

Peach Bottom Atomic Power Station

(Docket 50-171)

The Peach Bottom Atomic Power Station is to consist of a 115-Mw(t) helium-cooled graphite-moderated reactor. It will be built by the Philadelphia Electric Company on the Susquehanna River.[31] Following favorable review of the proposed site by the ACRS, as reported in the September 1960 issue of *Nuclear Safety*,[32] the Company applied for a construction permit in July 1960. The Division of Licensing and Regulation deferred consideration of this request until information could be submitted, evaluated, and approved concerning the description of the research, development, and testing programs for the fuel elements, control rods, control-rod drives, and fission-product-trapping system.[33] In the meantime, letters indicating intention to intervene at the eventual hearing on the construction permit have been received from Rep. James M. Quigley (D., Pa.), Lawrence W. Goldberg of Philadelphia, and C. L. Wilber, Secretary of Health of the State of Pennsylvania.

Pathfinder Atomic Power Plant

(Docket 50-130)

The Pathfinder Atomic Power Plant[34] is being constructed by the Northern States Power Company on the Big Sioux River, 3.5 miles northeast of Sioux Falls, S. Dak. The 203-Mw(t) reactor is to be a controlled-recirculation boiling-water system with nuclear superheat. Approval of construction was reviewed in the June and September 1960 issues of *Nuclear Safety*. In September 1960 the Company submitted a supplement[35] to its application which reflected the change of the boiler-region fuel cladding. This supplement was subsequently reviewed by the ACRS, which concluded in its letter of Nov. 5, 1960, that:[36]

> The Committee believes that the proposed substitution of Zircaloy-II for aluminum in the fuel elements of the boiling region of the core constitutes an acceptable change in the design of these elements.
>
> The Committee notes that there is no experience with the superheater fuel elements under the proposed operating conditions. Although we doubt that this feature of the reactor poses an undue hazard, this uncertainty should be resolved by adequate prior testing of the fuel elements. The Committee believes that operation of the superheater elements can be carried out without undue hazard to the public if increases in power level are made stepwise and are supplemented by a concurrent fuel element testing program.
>
> The Committee urges that the significant problems of the superheater design be resolved by the results of the Division of Reactor Development program on the superheater concept (such as BORAX V and loop tests), before a high power load is placed on the superheater of this reactor.

Saxton Nuclear Experimental Corporation

(Docket 50-146)

Approval for the construction of a 20-Mw(t) pressurized-water reactor by the Saxton Nuclear Experimental Corporation was discussed in the June 1960 issue of *Nuclear Safety*.[37] In August 1960 the Company submitted an amendment[38] to its application for a construction permit pertaining to a change in design of the reactor pressure vessel from a solid-wall construction to a multilayer construction. The ACRS subsequently concluded[39] that "the change to a multilayered pressure vessel designed and constructed specifically for the Saxton vessel will not introduce any additional hazard to the health and safety of the public."

Rural Cooperative Power Association

(Docket PP-1)

The Elk River reactor is a 58-Mw(t) boiling-water power reactor with natural circulation to be operated by the Rural Cooperative Power Association. As previously reported,[40] a public hearing was held prior to the construction of this facility, following which the construction was authorized by the hearing examiner. The ACRS, at its November 1960 meeting, reviewed the proposed operation of the facility and concluded:[41]

> From previous reviews all safety matters pertinent to operation had been resolved except: the composition of the startup crew; radiation monitors in areas outside the containment; personnel entrance to the containment; operating instructions; pressure coefficient; control rod operation; reactivity effect of fuel element motion; and repair of reactor pressure vessel nozzles.
>
> As a result of this review, the Advisory Committee on Reactor Safeguards concludes that all issues have been satisfactorily resolved. The Committee believes that this reactor can be operated without undue hazard to the health and safety of the public.

A public hearing was subsequently scheduled for December 1960 to consider whether authorization for the operation will be given.

Big Rock Point Nuclear Power Plant

(Docket 50-155)

The Big Rock Point reactor will be a forced-circulation boiling-water reactor having a maximum generating capacity of 75 Mw(e) [240 Mw(t)]. It is being constructed by the Consumers Power Company.[42] Following issuance of the construction permit, as previously discussed,[43] a number of design changes were proposed and reviewed by the ACRS, including modifications of the reactor containment, control-rod design, core instrumentation, and core power density. The ACRS concluded that, in its opinion:[44]

> A. The proposed specifications for the containment are satisfactory with respect to possible exposures to the health and safety of the public;
>
> B. The control rods and the in-core instrumentation will perform as well as or better than their counterparts now operating in the Dresden Reactor;
>
> C. The proposed program of approach to the power density of 45 kw/liter can be carried out safely; and
>
> D. A boiling water reactor of the proposed type as outlined in the application with Amendment No. 2 can be constructed at the proposed site with reasonable assurance that it may be operated as proposed without undue risk to the health and safety of the public.

A hearing was subsequently set[45] to consider amending the construction permit to include these changes and technical specifications and to consider findings with respect to the proposed stepwise approach to rated power.

Enrico Fermi Atomic Power Plant

(Docket F-16)

The Enrico Fermi reactor is a 300-Mw(t) fast breeder reactor which the Power Reactor Development Company is constructing at Lagoona Beach, Mich. As previously discussed,[46] the intervening AFL-CIO labor unions had appealed to the courts, stating that the AEC had violated its own rules in issuing the construction license for this facility, and the unions' appeal had been sustained by the U. S. Court of Appeals. When this court declined to review the decision, the Company placed the case before the U. S. Supreme Court.

In an action[47] which parallels that of the Company, government attorneys have asked the U. S. Supreme Court to rule that "existing law does not preclude the AEC from approving a site for an atomic power reactor that is close to a populated area—based on Commission findings that the site is suitable for a reactor of the general type and that there is reasonable assurance its operation there would not cause undue risk to

the public health and safety . . . ." The union brief and rebuttals by both the Company and the Justice Department, jointly with the AEC, were subsequently filed,[48,49] and the U. S. Supreme Court in November 1960 agreed[50] to review the case but limited consideration to the following two issues:

> 1. Whether the Atomic Energy Act precludes the Commission from approving a site for a power reactor that is close to a populated area unless the Commission finds that there are "compelling reasons" for such location.
>
> 2. Whether the Act requires the Commission, in granting a permit for construction of a power reactor, to make the same findings with respect to the safety of its operation as are required before it grants an operating license.

The U. S. Supreme Court said it would not consider (1) whether respondent unions were persons aggrieved by the grant of a construction permit, (2) whether the alleged injury would result from operation and not from construction of the reactor, and (3) whether a further AEC order was required before the reactor could be operated.

Prior to this, the Power Reactor Development Company had requested that its construction permit be amended to extend the completion date one year (to Dec. 15, 1961) because of developmental problems that were attributable to the experimental nature of the plant and were beyond the applicant's control. This request was approved over the opposition of the unions.[50]

Dresden Nuclear Power Station

(Docket 50-10)

The Dresden Nuclear Power Station of the Commonwealth Edison Company was shut down in November 1960 when a control rod separated from its drive mechanism following a month of full-power operation. As noted in the December 1960 issue of *Nuclear Safety*,[51] the reactor had been previously operated at power levels of up to 630 Mw(t), and, as specified in its interim license, the Company had applied for another hearing to consider the issuance of a 40-year operating license. The operating experience up to that time and a report entitled "Full Rated Power Test Operation of the Dresden Reactor," issued Aug. 8, 1960, were reviewed by the ACRS at its September meeting. The ACRS concluded that:[52]

> Based on the information contained in the referenced report and that presented orally at the meeting on September 22, it is the conclusion of the Advisory Committee on Reactor Safeguards that the Dresden Nuclear Power Plant is operating in accordance with prior expectations and that there is no reason to modify our opinion as set forth in our letter to you dated May 18, 1959, that this plant can be operated without undue risk to the health and safety of the public. We also see no reason to doubt that this will continue to be the case after the transfer of operating responsibility from the General Electric Company to the Commonwealth Edison Company, inasmuch as Commonwealth personnel have, in fact, already successfully taken the plant to full power operation many times and General Electric will continue to exercise some technical surveillance over the plant under a consulting arrangement with Commonwealth Edison.

At the subsequent public hearing, both the Company and the AEC witnesses testified[53] that Commonwealth Edison was qualified to assume operation of the plant under a 40-year license; the examiner concurred, and the license amendment was issued in mid-October 1960.[54] On Nov. 16, 1960, the AEC was notified:[55]

> Confirming conversation with Dr. Peter Norris, Dresden reactor was shut down yesterday when instrumentation indicated that a poison blade may have separated from its drive. In operating the drive after shutdown to check its condition, it appears that we have jammed the mechanism with the poison section three quarters out of the core. Now removing reactor lid to investigate further. Will not resume operation until trouble has been determined, reported and corrected. All other control rods appear to be in good operating condition.

The AEC immediately ordered that the period for its review of an "AEC's Examiner's decision authorizing full power operation of the Dresden plant to be extended from November 25 to December 16." It was also reported[55] that: "In its turn, Commonwealth Edison last week told the Commission that it will submit a further report on the matter and that the plant will not be operated until AEC has had an opportunity to evaluate the report." (W. B. Cottrell)

Safeguards Reports and Selected Reading

The recently issued safeguards reports and selected literature pertaining to hazards and

safety of reactors are listed below for reference. Because of the similarity of many reactors (in particular the research reactors), the list is not intended to be all-inclusive.

1. N. G. Wittenbrock, ed., Plutonium Recycle Test Reactor Final Safeguards Analysis. Supplement 1: Additional Studies, USAEC Report HW-61236(Suppl. 1), Hanford Atomic Products Operation, Apr. 15, 1960.

2. Bureau of Ships, Nuclear Propulsion Plant Safety Manual, Report NAVSHIPS-389-0197, January 1960. (Classified)

3. T. F. Davis, comp., Radioactive Waste Processing and Disposal: A Bibliography of Selected Classified Report Literature, USAEC Report TID-3069, October 1959. (Classified)

4. Aerojet-General Nucleonics, Army Gas-Cooled Reactor Systems Program, GCRE-I Hazards Summary Report, USAEC Report IDO-28506, Addendum II, February 1960, and Addendum III, May 1960.

5. C. C. Beusman, comp., Preliminary Safety Analysis of the Sodium-Deuterium Reactor (SDR), USAEC Report NDA-84-14, Nuclear Development Corp. of America, Mar. 6, 1959.

6. L. H. Jones and R. S. Kern, ETR Hazards Survey for GEANP-4, 5, and 6, USAEC Report IDO-16493, Phillips Petroleum Co., Oct. 5, 1958.

7. R. J. Smith, comp., Reactor Safety: A Bibliography of Classified Report Literature, USAEC Report TID-3074, October 1959. (Classified)

8. R. J. Smith, comp., Reactor Safety: A Bibliography of Selected Classified Report Literature Released to the Civilian Application Program, USAEC Report TID-3215, October 1959. (Classified)

9. E. O. Swickard, comp., Los Alamos Molten Plutonium Reactor Experiment (LAMPRE) Hazards Report, USAEC Report LA-2327, Los Alamos Scientific Laboratory, June 1959.

10. J. D. Simpson, Summary Hazards Evaluation of the SMR Facility, USAEC Report APEX-536, General Electric Co., Aircraft Nuclear Propulsion Dept., December 1959.

11. H. Dopchie and J. Planquart, Safety of the BR-2, the Belgium Materials Testing Reactor, in Symposium on High-Flux Materials Testing Reactors, Held in Brussels, Belgium, September 21–26, 1959, USAEC Report TID-7584, pp. 134–147, June 1960.

12. Henry Rosenthal, PMZ-1 Hazards Summary Report, USAEC Report MND-M-1854, Martin Co., Nuclear Div., December 1959.

13. R. A. Egen et al., Hazards Summary Report for the VMR Critical-Assembly Experiments, USAEC Report BMI-1445, Battelle Memorial Institute, June 10, 1960.

14. Puerto Rico Water Resources Authority and General Nuclear Engineering Corp., Boiling Nuclear Superheater (BONUS) Power Station: Preliminary Design Study and Hazards Summary Report, USAEC Report TID-8524, June 1960.

15. The Safest Industry, *Engineering*, 189 (4909): 691–692 (May 1960).

References

1. R. W. Dickinson and L. E. Glasgow, ASME Paper 59-A-192. (Unpublished)
2. Sodium Reactor Experiment Incident, *Nuclear Safety*, 1(3): 73–75 (March 1960).
3. A. A. Jarrett, SRE Fuel-Element Damage, USAEC Report NAA-SR-4488, Atomics International, Nov. 30, 1959.
4. C. Starr and R. W. Dickinson, *Sodium Graphite Reactors*, Addison-Wesley Publishing Company, Inc., Reading, Mass., 1958.
5. R. W. Dickinson, Coolant Block Damages SRE Fuel, *Nucleonics*, 18(1): 107–111 (1960).
6. K. W. Foster, Thermal Performance of the SRE Main Intermediate Heat Exchanger, USAEC Report NAA-SR-3775, Atomics International, June 15, 1960.
7. A. C. Werden, Jr., Southern California Edison Company Steam Electric Plant Operation, in Proceedings of the SRE-OMRE Forum, Held at Los Angeles, California, February 12 and 13, 1958, USAEC Report TID-7553, pp. 35–56.
8. Atomics International. (Unpublished)
9. F. A. Fillmore, Analysis of SRE Power Excursion, Report NAA-SR-5898 (to be published).
10. W. J. Anderson and G. V. Sneesby, Carburization of Austenitic Stainless Steel in Liquid Sodium, USAEC Report NAA-SR-5282, Atomics International, Sept. 1, 1960.
11. R. J. Hall and F. C. Six, An Integrated Turbine-Reactor Control System for the SRE, conference paper to be presented at AIEE Winter Meeting, February 1961.
12. C. W. Griffin et al., Measurement of the SRE Power Coefficients and Reactor Parameters Utilizing the Oscillation Techniques, USAEC Report NAA-SR-3763, Atomics International, Nov. 1, 1960.
13. R. W. Woodruff, Reactivity Absorbed by Xenon-135 in the SRE, USAEC Report NAA-SR-4328, Atomics International, Jan. 28, 1960.
14. J. G. Lundholm et al., Measurement of Zero Power Frequency Response in the SRE, USAEC

Report NAA-SR-3762, Atomics International, Nov. 1, 1960.

15. L. Kornblith et al., The General Electric Developmental Boiling Water Reactor, Final Hazards Summary Report, Report SG-VAL-2(3rd Ed.), General Electric Co., Vallecitos Atomic Laboratory, Nov. 30, 1959.
16. Action on Reactor Projects by Licensing and Regulating Bodies, *Nuclear Safety*, 2(2): 81–89 (December 1960).
17. *Atomic Energy Clearing House*, 6(34): 19 (Aug. 22, 1960).
18. *Atomic Energy Clearing House*, 6(45): 17 (Nov. 7, 1960).
19. Pacific Gas and Electric Co., Preliminary Hazards Summary Report, Humboldt Bay Power Plant Unit No. 3, Report NP-7512, Apr. 15, 1959, and Amendments 1–6, May 6, 1960.
20. Action on Reactor Projects by Licensing and Regulating Bodies, *Nuclear Safety*, 2(2): 87–88 (December 1960).
21. *Atomic Energy Clearing House*, 6(35): 19 (Aug. 29, 1960).
22. *Atomic Energy Clearing House*, 6(34): 6 (Aug. 22, 1960).
23. *Atomic Energy Clearing House*, 6(47): 14 (Nov. 21, 1960).
24. National Aeronautics and Space Administration Final Hazards Summary Report for NASA Plum Brook Reactor Facility, Parts I–III, December 1959.
25. Action on Reactor Projects by Licensing and Review Bodies, *Nuclear Safety*, 2(1): 111–112 (September 1960).
26. *Atomic Energy Clearing House*, 6(43): 16 (Oct. 24, 1960).
27. Letter from ACRS Chairman to Hon. John A. McCone, Nov. 7, 1960, subject: Report on NASA Plum Brook Facility.
28. Action on Reactor Projects by Licensing and Regulating Bodies, *Nuclear Safety*, 2(2): 86–87 (December 1960).
29. *Atomic Energy Clearing House*, 6(34): 15 (Aug. 22, 1960).
30. *Atomic Energy Clearing House*, 6(47): 14 (Nov. 21, 1960).
31. Philadelphia Electric Co., Site Evaluation Report for the Peach Bottom Atomic Power Station, Parts I and II, December 1959.
32. Action on Reactor Projects by Licensing and Review Bodies, *Nuclear Safety*, 2(1): 107–115 (September 1960).
33. *Atomic Energy Clearing House*, 6(39): 6 (Sept. 26, 1960).
34. Northern States Power Co., Mar. 10, 1959. (Unpublished)
35. *Atomic Energy Clearing House*, 6(37): 4 (Sept. 12, 1960).
36. Letter from ACRS Chairman to Hon. John A. McCone. Nov. 5, 1960, subject: Report on Northern States Power Company (Pathfinder).
37. Action on Reactor Projects by Licensing and Regulating Bodies, *Nuclear Safety*, 1(4): 90–94 (June 1960).
38. *Atomic Energy Clearing House*, 6(37): 5 (Sept. 12, 1960).
39. Letter from ACRS Chairman to Hon. John A. McCone, Sept. 26, 1960, subject: Saxton Nuclear Experimental Corporation.
40. Hearings on Reactor Projects, *Nuclear Safety*, 1(3): 81–85 (March 1960).
41. Letter from ACRS Chairman to Hon. John A. McCone, Nov. 5, 1960, subject: Report on Elk River Reactor—RCPA.
42. Consumers Power Co., Application to USAEC for Reactor Construction Permit and Operating License. Part B. Preliminary Hazards Summary Report, Report NP-8329, 1959.
43. Action on Reactor Projects by Licensing and Review Bodies, *Nuclear Safety*, 2(1): 107–115 (September 1960).
44. Letter from ACRS Chairman to Hon. John A. McCone, Nov. 5, 1960, subject: Report on Big Rock Plant—Consumers Power Company.
45. *Atomic Energy Clearing House*, 6(46): 2 (Nov. 14, 1960).
46. Action on Reactor Projects by Licensing and Regulating Bodies, *Nuclear Safety*, 2(2): 81–88 (December 1960).
47. *Atomic Energy Clearing House*, 6(40): 32 (Oct. 3, 1960).
48. *Atomic Energy Clearing House*, 6(44): 18 (Oct. 31, 1960).
49. *Atomic Energy Clearing House*, 6(45): 3 (Nov. 14, 1960).
50. *Atomic Energy Clearing House*, 6(47): 1 (Nov. 21, 1960).
51. Action on Reactor Projects by Licensing and Regulating Bodies, *Nuclear Safety*, 2(2): 86 (December 1960).
52. Letter from ACRS Chairman to Hon. John A. McCone, Sept. 24, 1960, subject: Dresden Atomic Power Station—Commonwealth Edison Company.
53. *Atomic Energy Clearing House*, 6(40): 20 (Oct. 3, 1960).
54. *Atomic Energy Clearing House*, 6(43): 10 (Oct. 24, 1960).
55. *Atomic Energy Clearing House*, 6(48): 1 (Nov. 28, 1960).